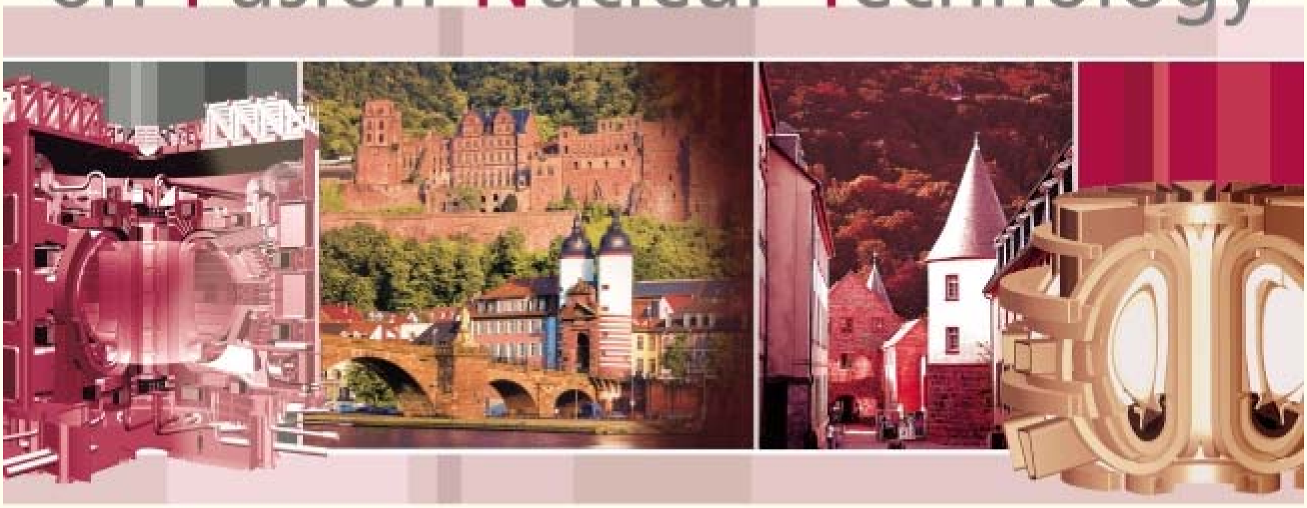
8th International Symposium on Fusion Nuclear Technology



Heidelberg | Germany | Sept. 30 - Oct. 5, 2007

Programme

Time	Sun, Sep 30	Mon, Oct 1	Tue, Oct 2	Wed, Oct 3	Thu, Oct 4	Fri, Oct 5
8:00-8:30		Registration				
8:30-10:30		Opening Opening Talk Keynote K-0001	Parallel Session S2	Keynote K-0002 Plenary P2	Plenary P3 Parallel Session S5	Plenary P5 Parallel Session S6
10:30-11:00		Break				
11:00-13:00		Plenary P1	Poster Session PS2	Parallel Session S3	Poster Session PS4	Panel Discussion
13:00-13:30		Lunchtime	Lunchtime	Lunchtime	Lunchtime	Lunchtime
13:30-14:30			Technical Tour (FZK)			
14:30-16:30		Parallel Session S1		Parallel Session S4	Keynote K-0003 Plenary P4	Parallel Session S7
16:30-17:00		Break		Break	Break	Closing Session
17:00-19:00	Registration & Welcome Reception	Poster Session PS1	Beergarden Party (FZK)	Poster Session PS3	Poster Session PS5	
19:00-19:30						
19:30-20:00		Welcome in Heidelberg			Conference Banquet	
20:00-22:00						
22:00-23:00						

Programme

Sunday, September 30

Time	Foyer
17:00-20:00	Registration & Welcome Reception

Programme

Monday, October 1

Time	Grand Hall	Ballroom	Chamber Music Hall	Foyer	
08:00-08:30				Registration	
08:30-09:00	**Opening** Dr. Janeschitz Dr. Gerner Prof. Maschuw Prof. Tanaka				
09:00-09:45	**Opening Talk** Prof. Llewellyn-Smith				
09:45-10:30	**Keynote K-0001** Iter Construction Norbert Holtkamp				
10:30-11:00	Break				
11:00-13:00	**Plenary P1** Fusion and Satellite Tokamaks 11:00 - P1-0001 Neil Mitchell 11:30 - P1-0002 Kimihiro Ioki 12:00 - P1-0003 Jerry Sovka 12:30 - P1-0004 Yoshikazu Okumura				
13:00-14:30	Lunchtime				
14:30-15:40	**Session S1** New Machines 14:30 - S1-0001 Makoto Matsukawa 15:00 - S1-0002 Yeong-Kook Oh	**Session S1** Repair and Maintenance (including HC) 14:30 - S1-0006 Alessandro Tesini 15:00 - S1-0007 Laurent Gargiulo 15:20 - S1-0008 Nobukazu Takeda			

15:40-15:50	15:25 - S1-0003 Jiangang Li	**Session S1** TBM Integration in ITER 15:40 - S1-0009 Valery Chuyanov 16:10 - S1-0010 Alfredo Portone			
15:50-16:30	**Session S1** First Wall Techn. & HHF Comp I 15:50 - S1-0004 Patrick Lorenzetto 16:10 - S1-0005 Ahmed Hassanein				
16:30-17:00	Break				
17:00-19:00			**Poster Session PS1** FW Technologie and HHFC ITER		
19:00-19:30					
19:30-22:00					Welcome in Heidelberg

Programme

Tuesday, October 2

Time	Grand Hall	Ballroom	Chamber Music Hall	
08:30-10:30	**Session S2** Models&Experiments I 08:30 - S2-0001 Paul Wilson 09:00 - S2-0002 P. Batistoni 09:25 - S2-0003 Akira Kohyama 09:50 - S2-0004 Haileyesus Tsige-Tamirat 10:10 - S2-0011 Chikara Konno	**Session S2** TBM Programme 08:30 - S2-0006 Lorenzo Virgilio Boccaccini 09:00 - S2-0007 C. p.c. Wong 09:30 - S2-0008 Kaiming Feng 09:50 - S2-0009 Jean-Francois Salavy 10:10 - S2-0010 Seungyon Cho		
10:30-11:00	Break			
11:00-13:00			**Poster Session PS2** Models&Experiments TBM Programme Satellite Tokamaks	
13:00-13:30	Lunchtime			
13:30-14:15				
14:15-17:00				Technical Tour in FZK
17:00-20:00				Beergarden Party at FZK

Programme

Wednesday, October 3

Time	Grand Hall	Ballroom	Chamber Music Hall
08:30-09:15	**Keynote K-0002** DEMO strategies Minh Quang Tran		
09:15-10:30	**Plenary P2** Fusion Beyond ITER 09:15 - P2-0001 David Maisonnier 09:40 - P2-0002 Farrokh Najmabadi 10:05 - P2-0003 Prof. Satoru Tanaka		
10:30-11:00	Break		
11:00-12:10	**Session S3** NSD and Reactor Studies 11:00 - S3-0001 Boris Kolbasov 11:25 - S3-0002 Chuanhong Pan 11:50 - S3-0003 Myeun Kwon 12:15 - S3-0004 Radhakrishnan Srinivasan 12:40 - S3-0005 Bong Guen Hong	**Session S3** First Wall Techn. & HHF Comp II 11:00 - S3-0006 Prachai Norajitra 11:25 - S3-0007 Volker Philipps 11:50 - S3-0008 J.g. Van Der Laan	
12:10-13:00		**Session S3** ICF Studies and Technologies 12:10 - S3-0009 Wayne Meier 12:40 - S3-0010 Mohamed E. Sawan	
13:00-14:30	Lunchtime		
14:30-16:00	**Session S4** Blanket Technology I 14:30 - S4-0001 Hongli Chen 14:55 - S4-0002 Thomas Ihli 15:20 - S4-0003 Neil Morley 15:50 - S4-0008 Rajendra Kumar Ellappan 16:10 - S4-0009 Hiroyasu Tanigawa	**Session S4** FNT Contributions to Other Fields of Sc.&Tech. 14:30 - S4-0005 Edgar Bogusch 15:00 - S4-0006 Hiroshi Horiike 15:30 - S4-0007 Tom Mehlhorn	
16:00-16:30		**Session S4** Safety Issues & Waste Management 16:00 - S4-0004 Laila El-Guebaly	

16:30-17:00	Break	
17:00-19:00		**Poster Session PS3** NSD and Reactor Studies Blanket Technology FNT Contributions to Other Fields of Sc.&Tech. ICF Studies and Technologies

Programme

Thursday, October 4

Time	Grand Hall	Ballroom	Chamber Music Hall	
08:30-09:30	**Plenary P3** Fuel Cycle and Plasma Burning 08:30 - P3-0001 Satoshi Konishi 09:00 - P3-0002 Kenichi Kurihara			
09:30-09:50	**Session S5** Fuel Cycle and T Processing I 09:30 - S5-0001 David Murdoch 09:50 - S5-0002 Takumi Hayashi 10:10 - S5-0003 Deli Luo	**Session S5** Burning Plasma Control and Operation 09:30 - S5-0004 Roger Raman		
09:50-10:30		**Session S5** Maintenance&Safety II 09:50 - S5-0005 Bernhard Haist 10:10 - S5-0006 Sandrine Rosanvallon		
10:30-11:00	Break			
11:00-13:00			**Poster Session PS4** Fuel Cycle and T Processing Burning Plasma Control and Operation Safety Issues and Waste Management Non Tokamak Machines	
13:00-14:30	Lunchtime			
14:30-15:15	**Keynote K-0003** Materials Anton Moeslang			
15:15-	**Plenary P4** Material Engineering and IFMIF 15:15 - P4-0001 Pascal Garin			

16:30	15:40 - P4-0002 K. Bhanu Sankara Bhanu 16:05 - P4-0003 Tatsuo Shikama		
16:30-17:00	Break		
17:00-19:00		**Poster Session PS5** Material Engineering IFMIF	
19:00-19:30			
19:30-23:00			Conference Dinner

Programme

Friday, October 5

Time	Grand Hall	Ballroom
08:30-09:45	**Plenary P5** Non Tokamak Reactor Studies 08:30 - P5-0001 Rene Raffray 08:55 - P5-0002 Osamu Motojima 09:20 - P5-0003 Robert Wolf	
09:45-10:25	**Session S6** Material Engineering for FNT 09:45 - S6-0001 A-A. F. Tavassoli 10:10 - S6-0005 Qi Xu	**Session S6** Fuel Cycle and T Processing II 09:45 - S6-0003 Sergey Beloglazov 10:05 - S6-0004 Alexander Perevezentsev
10:25-10:30		
10:30-11:00	Break	
11:00-13:00	**Panel Discussion:** A fast track approach to DEMO	
13:00-14:30	Lunchtime	
14:30-14:50	**Session S7** IFMIF&Materials 14:30 - S7-0001 Alban Mosnier 15:00 - S7-0002 Hiroo Nakamura 15:25 - S7-0003 Anton Möslang 15:50 - S7-0004 Dieter Leichtle	**Session S7** Models&Experiments II 14:30 - S7-0006 Rosaria Villari
14:50-16:10		**Session S7** Blanket Technology II 14:50 - S7-0008 Takanori Hirose 15:10 - S7-0009 Tomoaki Hino 15:30 - S7-0010 Lida Magielsen 15:50 - S7-0011 Luis Sedano
16:10-16:40	Closing Session	

STATUS OF THE ITER CONSTRUCTION PREPARATION

Norbert Holtkamp **(a)**, Günter Janeschitz **(a)**

(a) Forschungszentrum Karlsruhe

ITER as an organization has been established officially on Nov 21st 2006. Together with the creation of this international body the participating countries and the ITER International Organization have committed to a construction schedule of about 10 years under a fixed budget. ITER for the first time should bring together reactor-grade plasma and current technology, in an attempt to see how a viable energy source can be built.
Apart from the scientific challenge, ITER will be the first mega-science project that is to be build under an "in-kind" arrangement in which contributions from the collaborating countries are given in terms of ready-to-install subsystems for the facility and only to a small extent in cash. The main engineering challenge is to turn the existing designs into procurement packages that can be executed within the countries on time, while ensuring an integrated design.
In addition maintaining some flexibility in the layout to respond to changes in understanding as the device operates is key to every scientific endeavour. The current ITER design was completed in 2001, and a number of changes have been proposed since then. A design review process is underway to address outstanding design issues, to identify any new ones, to integrate solutions, and to ensure that the schedule and objectives can be met. This involves the expertise of the ITER Project Team, along with experts from the participating countries, and will focus initially on long lead items and related basic systems to provide a framework for later procurements.
In addition to the above, the paper will address the status of the adaptation to the Cadarache site near Aix-en-Provence, the licensing process of the nuclear facility within France, and will show the construction progress.

OVERVIEW OF THE ITER MAGNET SYSTEM

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(a) ITER IT Cadarache, 13108 St. Paul lez Durance, France

As ITER starts the construction phase, the magnets are one of the items on the critical procurement path. The basic design accepted by the ITER participants dates from 2001. The first step in the release of the various procurement packages has been the completion of critical R&D to confirm the design performance as well as a thorough review into the design solutions being proposed (including changes introduced since 2001).

The baseline ITER design is by now quite well known, with 18 Toroidal Field (TF) D shaped coils storing 44GJ of magnetic energy at fields close to 12T, a Central Solenoid (CS) stack of 6 modules operating up to 13T and 6 large Poloidal Field (PF) coils at 6T. Correction of manufacturing and assembly errors in the magnet systems is provided by a set of 18 low field Correction Coils (CC). The coils use cable-in-conduit (CIC) conductor with both Nb3Sn and NbTi superconducting strands cooled by supercritical helium. The low temperature allows the cryogenic strength of structural steels to be exploited with primary stresses approaching 700MPa in compression. Operating voltages on the coils are in the 10kV range with insulation designed in the 20-30kV range to allow a good margin for possible fault conditions.

To allow a focused design review, the recent work has concentrated on design areas with controversial or novel features, or on components where performance verification is incomplete. This work is now nearing completion. Areas selected for review, sometimes without resulting in design changes, are
(i) the Nb3Sn conductor design and the superconducting performance degradation seen in some recent test samples;
(ii) the TF coil windings and the use of a winding configuration to provide insulation redundancy;
(iii) the magnet structures, the material requirements compared to the available manufacturing capacity and optimisation to reduce them;
(iv) the finalising of the structural design criteria with appropriate design margins against possible failure mechanisms;
(v) features that reduce the risk of the machine becoming inoperable through electrical failure.

The work, and the resulting final design, will be summarised in the paper

ITER VACUUM VESSEL, IN VESSEL COMPONENTS AND PLASMA FACING MATERIALS

Kimihiro Ioki **(a)**, M. Enoeda **(b)**, G. Federici **(c)**, B. C. Kim **(d)**, I. Mazul **(e)**, M. Merola **(a)**, M. Morimoto **(a)**, M. Pick **(a)**, V. Rozov **(a)**, S. Suzuki **(b)**, M. Ulrickson **(f)**, Yu. Utin **(a)**, X. Wang **(a)**, S. Wu **(g)**, J. Yu **(a)**

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The VV design is being developed in more details considering manufacturing and assembly methods, and cost. Incorporating manufacturing studies being performed in cooperation with parties, the regular VV sector design has been nearly finalized. Detailed design development of other sectors near NB ports at the equatorial level is also on progress. Design of the NB ports including duct liners under heat loads of the neutral beams has been developed.
Design of the in-wall shielding has been developed in more details considering the supporting structure and the assembly method. The ferromagnetic inserts have previously not been installed in the outboard midplane region due to irregularity caused by the tangential ports for NB injection. Due to this configuration, the maximum ripple is relatively large (~1 %) in a limited region of the plasma and the toroidal field flux lines fluctuate ~10 mm in the FW region. To avoid these problems, additional ferromagnetic inserts are to be installed in the equatorial port region.
Detailed studies were carried out on the ITER vacuum vessel to define appropriate codes and standards in the context of the ITER licensing in France. A set of draft documents regarding the ITER vacuum vessel structural code were prepared including an RCC-MR Addendum for the ITER VV with justified exceptions or modifications. The main deviation from the base Code is the extensive use of UT in lieu of radiography for the volumetric examination of all one-side access welds of the outer shell and field joint.
The procurement allocation of blanket modules among 6 parties was fixed and the blanket module design has progressed in cooperation with parties. Fabrication of mock-ups for prequalification testing is under way and the tests will be performed in 2007-2008. Development of new beryllium materials is progressing in China and Russia.
The ITER limiters will be installed in equatorial ports at two toroidal locations. The limiter plasma-facing surface protrudes ~8 cm from the FW during the start-up and shutdown phase. In the new limiter concept, the limiters are retracted by ~8 cm during the plasma flat top phase. This concept gives important advantages; (i) mitigation of the particle and heat loads due to disruptions, ELMs and blobs, (ii) improvement of the power coupling with the ICRH antenna in a protected position flush with the FW. There are no sliding supports inside the vacuum, to keep the reliability of the system. Driving mechanisms are located outside the vacuum boundary.
The divertor activities have progressed with the aim of launching the procurement according to the ITER project schedule. They include: (a) the consolidation of the design and manufacturing technologies for the plasma facing components (PFCs); (b) the prequalification programme by the parties prior to entering into the procurement phase, (c) the diagnostics integration into the divertor design, (d) the development of suitable acceptance criteria for the divertor PFCs including the required fabrication control methods; (e) the development of remote handling procedures for the first installation and for the

following replacements of the divertor cassettes.

P1-0003 (Invited)

ITER BUILDINGS, SITE LAYOUT AND SAFETY

Jerry Sovka and Jean-Philippe Girard **(a)**

(a) ITER Organisation, St. Paul-les-Durances, France

Previous generic designs for the ITER buildings, plant layout and safety features are being adapted to French regulatory requirements as the next steps in preparation for constructing this experimental fusion facility.

The characteristics of weather, environment, geology and seismic history, applicable to Cadarache, are being incorporated in the site layout and buildings` designs, as well as taking into account the special interests of local communities expressed through public hearings and debates. Currently, initial steps of site clearing are nearing completion, whilst temporary facilities are in preparation for subsequent construction activities such as earthworks, platform levelling, perimeter fences, and access control, as well as the introduction of essential services such as electric power, potable water, fire protection, temporary roads, sewage, and storm drains. As far as possible, areas are being set aside for the environmental protection of local trees, plants and insects.

ITER, with its seven international partners, is coordinating the supply of the equipment and facilities through negotiated procurement sharing agreements. Nevertheless, the design, fabrication, construction and installation will be required to meet European codes and standards as well as to follow French safety and licensing procedures.

This paper presents the current status of the building, site and licensing activites, in preparation for requesting the construction permit.

BROADER APPROACH TO FUSION ENERGY

Yoshikazu Okumura and Roberto Andreani **(a)**

(a) JAEA

The European Atomic Energy Community (EURATOM) and the Japanese government signed an agreement for "the Joint Implementation of the Broader Approach Activities in the Field of Fusion Energy Research" on 5th February 2007. The Agreement is expected to enter into force before the end of this year. This co-operation aims at complementing the ITER project and at an early realization of fusion energy by carrying out R&D and developing some advanced technologies for the future demonstration power reactor (DEMO). Three research projects will be undertaken in Japan under the framework of this agreement;

1. Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA).
IFMIF will allow testing and qualification of advanced materials in an environment similar to that of a future fusion power plant. The Engineering Validation and Engineering Design Activities aim at producing a detailed, complete and fully integrated engineering design of IFMIF.
2. International Fusion Energy Research Centre (IFERC).
The missions of the IFERC include the co-ordination of DEMO Design and R&D activities, large scale simulation activities of fusion plasmas by super-computer and remote experimentation activities to facilitate a broad participation of scientists into ITER experiments.
3. Satellite Tokamak Programme
The JT-60 tokamak will be upgraded to an advanced superconducting tokamak JT-60 SA, and be exploited as a "satellite" facility to ITER. The Satellite Tokamak Programme is expected to develop operating scenarios and address key physics issues for an efficient start up of ITER and in order to provide a continuous support to ITER experimentation and to advance research towards DEMO.

While the Satellite Tokamak Programme will be conducted at the existing JT-60 tokamak site in Naka, the IFMIF-EVEDA and the IFERC projects will be carried out at Rokkasho, Aomori. A new research site is being prepared for these two projects at the same site proposed for ITER. The construction of the buildings is to be started during this year and completed within two years. Preparatory work has been started such as the joint planning on the IFERC and the IFMIF-EVEDA, and the technical review of the conceptual design of JT-60SA. As soon as the Agreement will be in force, EURATOM and Japan will adopt the project plans for the three projects to be able to implement them without delay.

S1-0001 (Invited)

LATEST DESIGN STATUS OF JT-60SA TOKAMAK UNDER THE EU-JA BROADER APPROACH AGREEMENT

M. Matsukawa **(a)**, JT-60SA Design Team **(b)**

(a) Japan Atomic Energy Agency, Naka, Ibaraki 311-0193, Japan
*(b) **

JT-60SA is a Tokamak with a complete superconducting coils system to be built in the framework of the EU-JA Broader Approach Agreement, and it aims to contribute to the experimentation with ITER and to the DEMO reactor design. Its construction will start immediately after the final ratification of the Agreement by Japan, being expected mid 2007. The JT-60SA is designed by making maximum use of the existing facilities such as buildings, power supplies, plasma heating and current drive devices and diagnostics, and it will replace JT-60U in the torus hall. The maximum plasma current is 5.5 MA for low aspect ratio plasmas (Rp=3.06m, A=2.65, K95=1.76, d95=0.45) and 3.5 MA for ITER-shaped plasmas (Rp =3.15m, A=3.1, K95=1.69, d95=0.36). A plasma current flattop period of 100 s is expected in the standard discharge scenario under the maximum flux swing capability of ~40 Wb. The maximum plasma heating power of 41 MW with 100 s pulse duration is planned using 10 MW of N-NBI, 24 MW of, P-NBI and 7 MW of ECH. It must be noted that the vertical position of N-NBI beam line will be shifted 0.6 m below the equatorial plane of the machine for off-axis heating. P-NBI consists of 8 units of perpendicular injectors and 4 units of balanced tangential injectors. Each P-NBI injector has 2 MW heating power. The ECH system has two different frequencies of 110 GHz and 140 GHz. This improves the flexibility of operation for the NTM suppression, plasma initiation and current ramp-up assist, and wall cleaning.
In the superconducting TF coil design, NbTi conductor will be used with a copper/non copper ratio in the range of 1.6-1.9 to increase the thermal stability. In the CS design, using NbSn, highly manganised austenitic stainless steel JK2 is adopted for the conductor jacket material to mitigate the pre-compression stress of the tie-plates of CS stacks. In the EF coils, two kinds of NbTi conductor using the same superconducting strand are adopted to cope with the operation of the divertor coil at 6.2 T and of the outer ring coils at less than 5 T.
The annual DD neutron yield of JT-60SA is planned to reach 4x10^21 as the maximum, so that human access into the vacuum vessel is considerably restricted because the expected radiation dose level is 1-2 mSv/hr. Therefore, a remote handling system must be developed strongly coupled with the design of the in-vessel components. Mono-block type CFC armor will be adopted for the divertor target to handle the expected heat load of 15-20 MW/m², although the shape of the divertor modules has not been fixed yet while physics and implementation studies are still under way. The designs of cryo-plant, AC and DC power supplies, bio-shielding, water cooling system, N2 gas baking system, and the tokamak assembly time schedule will be presented at the conference.

*(JA side) M. Kikuchi, T. Fujii, T. Fujita, T. Hayashi, S. Higashijima, Y. Ikeda, S. Ishida, Y. Kamada, H. Kimura, K. Kizu, K. Kurihara, K. Masaki, M. Matsukawa, N. Miya, A. Sakasai, S. Sakurai, Y. Shibama, A. Sukegawa, M. Takechi, H. Tamai, K. Tsuchiya, K. Yoshida
*(EU side) R. Andreani, J. Alonso, J. Botija, A. Coletti, R. Coletti, P. Costa, A. Cucchiaro, P. Decool, A. Della Corte, A. Di Zenobio, N. Dolgetta, J-L. Duchateau, W.H. Fietz, E. Gaio, A. Grosman, O. Gruber, R. Heller, D. Henry, P. Hertout, J. Hourtoule, B. Lacroix, R. Magne, M. Medrano, F. Michel, L. Muzzi, S. Nicollet, L. Novello, L. Petrizzi, R. Piovan, A. Pizzuto, C. Portafaix, E. Rincon, S. Roccella, L. Semeraro, S. Turtù, J-M. Verder, S. Villari, L. Zani

S1-0002 (Invited)

COMPLETION OF THE KSTAR CONSTRUCTION AND ITS ROLE AS ITER PILOT DEVICE

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The Korea Superconducting Tokamak Advanced Research (KSTAR) device is under construction at the National Fusion Research Center (NFRC) with the mission of developing a steady-state capable advanced superconducting tokamak to establish the scientific and technological bases for an attractive fusion reactor.

The KSTAR project has been started from 1995 and the assembly milestone of the device is by the August 2007. Key achievements in the KSTAR construction are (i) fabrication of tokamak heavy structures, (ii) design and development of the Nb3Sn superconducting coils, (iii) accurate assembling of all components using special assembly tools, and (iv) design and fabrication of magnet structures to resist high magnetic force in TF and PF coils (v) developing ancillary systems including control system, magnet power supply, diagnostic systems, and heating systems.

The integrated commission and first plasma discharge are planned by 2008 to verify the machine construction quality and to estimate the operational capability of the device. During the integration commissioning, tokamak machine performance will be characterized such as superconducting coils and structures according to cool-down and current charging.

The goal of the KSTAR operation is to acquire the scientific and engineering knowledge of the high performance plasma confinement and long pulse operation. KSTAR device could have a role of ITER pilot device. The plasma operation control such as advanced tokamak operation with high beta plasma, steady-state operation, and resistive wall mode (RWM) operation with segmented in-vessel coils could be a good benchmark of the ITER operation. The operation experience of the KSTAR device could be also referred to ITER device engineering such as operation experience of Nb3Sn superconducting coils and low hybrid current drive (LHCD) system with same frequency of 5 GHz as that of ITER device.

In this paper, the status of the construction KSTAR device and operation plan as an ITER pilot device will be presented.

S1-0006 (Invited)

THE ITER REMOTE MAINTENANCE SYSTEM

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ITER is a joint international research and development project that aims to demonstrate the scientific and technological feasibility of fusion power.

As soon as the plasma operation begins using tritium, the replacement of the vacuum vessel internal components will need to be done with remote handling techniques. To accomplish these operations ITER has equipped itself with a Remote Maintenance System; this includes the Remote Handling equipment set and the Hot Cell facility. Both need to work in a cooperative way, with the aim of minimizing the machine shutdown periods and to maximize the machine availability.

The ITER Remote Handling equipment set is required to be available, robust, reliable and retrievable. The machine components, to be remotely handle-able, are required to be designed simply so as to ease their maintenance. The baseline ITER Remote Handling equipment is described.

The ITER Hot Cell Facility is required to provide a controlled and shielded area for the execution of repair operations (carried out using dedicated remote handling equipment) on those activated components which need to be returned to service, inside the vacuum vessel. The Hot Cell provides also the equipment and space for the processing and temporary storage of the operational and decommissioning radwaste. A conceptual ITER Hot Cell Facility is described.

DEVELOPMENT OF AN ITER RELEVANT INSPECTION ROBOT

Laurent Gargiulo **(a)**, Pascal Bayetti **(a)**, Jean-Jacques Cordier **(a)**, Jean-Pierre Friconneau **(b)**, Christian Grisolia **(a)**, Jean-Claude Hatchressian **(a)**, Delphine Keller **(b)**, Yann Perrot **(b)**

(a) Association Euratom-CEA, Département de Recherche sur la Fusion Contrôlée, CE-Cadarache, France
(b) CEA-LIST Robotics and Interactive Systems Unit, CE-Fontenay aux Roses, France

Robotic operations are one of the major maintenance challenges for ITER and future fusion reactors. In particular, in vessel inspection operations without loss of conditioning could be very useful.
Within this framework, the aim of the project called AIA (Articulated Inspection Arm) is to demonstrate the feasibility of a multi-purpose in-vessel Remote Handling inspection system using a long reach, limited payload carrier (up to 10 kg). It is composed of 5 segments with 11 degrees of freedom and a total range of 8 m.
The project is currently developed by the CEA within the European workprogramme. Its first in situ tests are planned this summer on the Tore Supra tokamak at Cadarache (France). They will validate chosen concepts for operations under ITER relevant vacuum and temperature conditions. After qualification, the arm will constitute a promising tool for generic application.
Several processes are already considered for ITER maintenance and will be demonstrated on the AIA robot carrier:
- The first embedded process is the viewing system. It is currently being manufactured and will allow for close visual inspection of the complex Plasma Facing Components (limiters, neutralisers, RF antennae, diagnostic windows, etc.).
- In situ localisation of leakage based on helium sniffer is also studied to improve maintenance operations.
- Finally the laser ablation system for PFC detritiation, also developed in CEA laboratories, is being fitted to be implanted into the robot and put into operation in Tore Supra.
This paper deals with the integration of the robot in the Tore Supra tokamak and the advances in the development of the listed processes. It also introduces the current test campaign aiming to qualify the robot performance and reliability under vacuum and temperature conditions.

S1-0008

DEVELOPMENT OF SIMULATOR FOR REMOTE HANDLING SYSTEM OF ITER BLANKET

Nobukazu Takeda **(a)**, Satoshi Kakudate **(a)**, Masataka Nakahira **(a)**, Yasuhiro Matsumoto **(a)**, Kiyoshi Shibanuma **(a)**, Alessandro Tesini **(b)**

(a) ITER Tokamak Device Group, Japan Atomic Energy Agency
(b) ITER Organization

The maintenance activity in the ITER has to be performed remotely because 14 MeV neutron caused by fusion reaction induces activation of structural material and emission of gamma ray. In general, it is one of the most critical issues to avoid collision between the remote maintenance system and in-vessel components. Therefore, the visual information in the vacuum vessel is required strongly to understand arrangement of these devices and components. However, there is a limitation of arrangement of viewing cameras in the vessel because of high intensity of gamma ray. It is expected that enough numbers of cameras and lights are not available because of arrangement restriction. Furthermore, visibility of the interested area such as the contacting part is frequently disturbed by the devices and components, thus it is difficult to recognize relative position between the devices and components only by visual information even if enough cameras and lights are equipped. From these reasons, the simulator to recognize the positions of each devices and components is indispensable for remote handling systems in fusion reactors.

The authors have been developed a simulator for the remote maintenance system of the ITER blanket using a general 3D robot simulation software "ENVISION". The simulator is connected to the control system of the manipulator which was developed as a part of the blanket maintenance system in the EDA and can reconstruct the positions of the manipulator and the blanket module using the position data of the motors through the LAN. In addition, it can provide virtual visual information, such as the connecting operation behind the blanket module with making the module transparent on the screen. It can be used also for checking the maintenance sequence before the actual operation.

The developed simulator will be modified further adding other necessary functions and finally completed as a prototype of the actual simulator for the blanket remote handling system which will be procured as a part of in-kind contribution.

S1-0003 (Invited)

EAST AND ITS TECHNICAL PROGRAM IN PREPARATION TO ITER

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EAST is the first Tokamak using superconducting magnet technology similar to that required in ITER. The successful construction and commissioning of EAST could provide many useful experiences for ITER. EAST got its first plasma on Sept. 26, 2006. Experiments have been carried out during last October and this January. Up to 500kA divertor plasma has been obtained. Up/Down single null, double null configurations have been obtained with elongation close to 2 and elongation over 0.6. Plasma duration was close to 10s. The success in achieving various shaped diverted plasma confirms capability of the superconducting poloidal magnets and plasma control algorithm with the EAST new features.

Efforts have been made for machine safety, reliability and capacity during commissioning. All design parameters of machine reached their full values, such as toroidal field 3.5T, 20kA/s PF coil ramping rate, in the end of commissioning. Of particular interest have been focused on the operational experience with quench detection systems, reliable interlock and safety system, the plasma control capability of the superconducting PF coils and the use of HTc current leads.

The new machine shows its unique features during experiments, which are well suited to answer a number of important issues for ITER operations and developments towards DEMO. Plasma initiation, ramp up and control with constraints of superconducting coils. Very low plasma ramp rate of 0.1MA/s during start up phase have been obtained with assistant of LHW on a boronized wall condition. Effects of AC losses and disruptions on the superconducting systems have been evaluated during plasma discharges. Two wall conditioning techniques, GDC and ICR, have been used and compared. ICR technique has been extensively used for wall cleaning, recycling control, and boronization with very wide operation pressure (1x10-4Pa-5Pa).

Further developments of EAST hardware will make more contribution for ITER construction and operation, such as investigation of particle inventory with various divertor configurations, graphite walls and development of hydrogen removal techniques, development of high performance operation in steady state condition in near future.

THE INTEGRATION OF TBM SYSTEMS IN ITER

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Testing of breeding blanket modules (TBMs) is one of the ITER goals foreseen from the very beginning of the ITER Project. At the same time formal arrangements for the testing have not been defined in the ITER Implementation agreement and are now under consideration by ITER parties. This paper does not consider these arrangements and reports only on technical aspects of the TBMs testing.

Starting from the ITER 2001 Final Report, the broadening of ITER partnership from 3 to 7 Parties, the interest of all the Parties to participate in the TBM testing and the shift of technical interests to helium cooling of the TBMs have created additional requirements in relation to the integration of TBMs systems in ITER.

Six half-port TBMs and associated systems are expected to be tested simultaneously in the three available Test Ports. This paper presents an initial assessment of the TBM and ITER interface requirements that will need immediate attention.

Four areas of interface were identified. The first area is the port cell interface area, including components like the port plug frame, backside shield, dummy plug, dummy TBM and corresponding tools needed for the TBMs maintenance and replacement. The second area is the hot cell, including the needed additional hardware for the service of TBMs, additional remote handling tools, and additional building space needed for the maintenance of the TBM ancillary equipment and the corresponding testing utilities and tools. The third area is the tokamak cooling water system (TCWS) with the need to accommodate six TBM heat transfer systems, each with a footprint of 57 m2. The fourth area of interface is the tritium plant.

All key facilities and building areas were identified, including the needed control room space for the 7 ITER parties. High pressure pipes connecting the port cell and TCWS area are also included. In all these areas modifications in the current ITER design are needed to accommodate the TMB testing. These changes must be incorporated in the new ITER baseline design which is now under preparation.

THE ITER TF COIL RIPPLE: EVALUATION OF RIPPLE ATTENUATION USING FE INSERT AND OF RIPPLE ENHANCEMENT PRODUCED BY TBM

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The ITER Toroidal Field Coil (TFC) system is made of 18 D shaped coils spaced by 20° in toroidal angle, this discontinuity can cause significant losses in the confinement of high energy particles (a-particles or high-energy ions from neutral beam injectors) due to their trapping inside the "ripple" valleys; the toroidal ripple of the flux surfaces in contact with the First Wall (FW) produces unwanted peaking in the heat loads on the FW itself. Due to these reasons an accurate evaluation of the Toroidal Field Ripple (TFR) in various operation conditions has been performed. To this end, various Finite Element Models (FEM), using the ANSYS code, have been developed. To produce regular field mapping, these models make use only of structured meshes that allow high filed precision and a very regular spacing of the model elements. The mapping has been extended to all the region internal to the FW including the FW itself. The FEM takes into account the real 3-D shape of the TFC. The FEM model of TFC is made of three nested D shaped coils capable of reproducing with high accuracy the real geometry of the TFC.

The value found of the TFR has confirmed the need of introducing some correcting elements.

The benefit of introducing Fe inserts between the two vessel shells at the outboard has been tested. It has been shown that excluding the inserts from the equatorial region, as it was made in all the previous works on this subject, does not allow any significant benefit, it has instead proven that a ripple reduction up to a factor 3 or more could be obtained including this region.

The poloidal field produces a misalignment between the magnetization of Fe insert, in stationary conditions aligned to the resultant field, and the toroidal field. The effect of this misalignment on the ripple correction produced by the inserts has been checked and was proven to be negligible.

The possible ripple over-compensation during plasma reduced scenario (at halved toroidal field) has been analyzed.

At the end, the field perturbation introduced by the presence of a Test Blanket Module (TBM) for DEMO in the equatorial port has been analyzed. It has been shown that the TMB (made of about 2.7 tons of EUFER and with saturated magnetization about 1.9 T) introduces a very large field perturbation: about three times the uncorrected and ten times the TFR corrected with the inserts.

In order to allow the analyses of the particle losses and of the heat loads, a detailed ripple map of the TFR has been produced for the whole region inside the FW and for all the main cases that have been analyzed: a) without inserts and without TBM, b) with inserts and without TBM, c) without inserts and with TBM, d) with inserts and with TBM. The relative precision in the error field obtained in these analyses is better than 1%.

S1-0004

STATUS OF THE EU R&D PROGRAMME ON THE BLANKET-SHIELD MODULES FOR ITER

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The ITER Blanket-shield concept is a modular configuration mechanically attached to the vacuum vessel. The modules consist of a water-cooled 316L(N)-IG Stainless Steel (SS) Shield Block and separable First Wall (FW) panels mechanically attached to the Shield Block. The FW panels are made from a bi-metallic structure with a 316L(N)-IG SS backing plate and a Copper (Cu) alloy heat sink layer, with Beryllium (Be) tiles as plasma facing material joined to the Cu alloy heat sink layer.
A Research and Development programme for the ITER Blanket-shield modules has been implemented in Europe to provide input for the design and the manufacture of the full-scale production components. It involves in particular the fabrication and testing of mock-ups and full-scale prototypes of Shield blocks and FW panels.
Two methods have been considered in Europe for the manufacture of the Shield blocks. The first method is based on a conventional welded fabrication technique while the second method uses a more advanced technique based on Hot Isostatic Pressing (HIPping) of 316L(N)-IG SS powder and 316L(N)-IG SS solid parts. A full-scale Shield prototype has recently been manufactured to demonstrate the feasibility of this second method.
Two methods have also been considered in Europe for the manufacture of the bi-metallic structure of the FW panels: solid and powder HIPping. Beryllium (Be) tiles are then joined by HIPping or brazing. Three full-scale FW panel prototypes have been completed and two are under manufacture.
This paper will present the latest developments of this R&D programme. In particular, it will report the latest results of the Shield fabrication development programme with the manufacture of the full scale Shield prototype. It will also report the latest results of high heat flux and thermal fatigue tests of FW mock-ups. It will describe the preparation of irradiation experiments of Be coated FW mock-ups. Finally, it will present a possible FW qualification programme to be implemented by the contributing Participant Teams prior to the start of the procurement of the blanket modules for ITER.

S1-0005

VERTICAL DISPLACEMENT EVENTS: A SERIOUS CONCERN IN FUTURE ITER OPERATION

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The strongly elongated plasma configuration in ITER-like devices is vertically unstable unless an active control feedback at the vertical position is applied. A malfunction of this feedback system for variety of reasons can lead to a rapid plasma vertical displacement at full plasma current. As the plasma contacts the top or bottom of the vacuum vessel, the current is rapidly forced to zero, similar to the behavior of the plasma after the thermal quench of a disruption. This phenomenon constitutes the vertical displacement events (VDE). This can result in melting and vaporization of the plasma-facing component (PFC) as well as melting of the copper substrate and burnout of the coolant channels. The upgraded HEIGHTS simulation package is used to simulate in full 3D the response of an entire ITER module response to a VDE. The initial temperature distribution of the PFC and the bulk substrate prior to the VDE is calculated according to steady state heat flux, module design, and initial coolant temperature. The models used in the upgraded HEIGHTS were recently benchmarked against VDE simulation experiments using powerful electron beam and show an excellent agreement with the data. The surface temperature can then be very high and could result in significant melting of substrate copper and damage the coolant channels. In the case of Be surface, surface vaporization is quite high and will remove most incoming plasma power at typical ITER VDE condition. Therefore, the transmitted heat flux to the substrate and the coolant channels are low enough to cause any significant damage. However, if tungsten is exposed to the VDE the situation is quite different. No significant surface vaporization will occur at the tungsten surface thus, leaving the majority of the incident plasma power to be conducted to the copper substrate causing melting at the interface and burnout of coolant channel with serious implications on the integrity and subsequent performance of this module. The results are presented in full 3D with movies showing both the PFC and structural response of an entire ITER module.

RESULTS OF WATER CORROSION IN STATIC CELL TESTS REPRESENTING MULTI-METAL ASSEMBLIES IN THE HYDRAULIC CIRCUITS OF TORE SUPRA

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Tore supra (TS) has used from the beginning of operation in 1989 actively cooled plasma facing components. Since the operation and baking temperature of all in vessel components has been defined to be up to 230 °C at 40 bars, a special water chemistry of the cooling water plant was suggested in order to avoid eventual water leaks due to corrosion (general corrosion, galvanic corrosion, stress corrosion, etc.) at relative high temperatures and pressures in tubes, pipes, bellows, water boxes, coils, etc. From the beginning of TS operation, in vessel components (e.g. wall protection panels, limiters, ergodic divertor coils, neutralisers and diagnostics) represented a unique combination of metals in the hydraulic circuit mainly such as stainless steel, Inconel, CuCrZr, Nickel and Copper. These different materials were joined together by welding (St to St, Inconel to Inconel, CuCrZr to CuCrZr and CuCrZr to St-St via a Ni sleeve adapter), brazing (St-St to Cu and Cu-LSTP), friction (CuCrZr and Cu to St-St), explosion (CuCrZr to St-St) and memory metal junction (Cryofit to Cu - only test sample).
Following experiences obtained with steam generator tubes of nuclear power plants, a cooling water quality of AVT (all volatile treatment) has been defined based on demineralised water with adjustment of the pH value to about 9.0/ 7.0 (25 °C/ 200 °C) by addiction of ammoniac, and hydrazine in order to absorb oxygen dissolved in water. At that time, a simplified water corrosion test program has been performed using static (no circulation) test cell samples made of above mentioned TS metal combinations. All test cell samples, prepared and filled with AVT water, were performed at 280 °C and 65 bars in an autoclave during 3000 hours. The test cell water temperature has been chosen to be sufficient above the TS component working temperature, in order to accelerate an eventual corrosion process.
Generally all above mentioned metal combinations survived the test campaign without stress corrosion cracking, with the exception of the memory metal junction (creep in Cu) and the bellows made of St-St 316L and Inconel 625 while 316 Ti bellows survived. In contrary to the vacuum brazed Cu-LSTP to St-St samples, some of flame brazed Cu to St-St samples failed either in the braze joint or in the copper structure itself. For comparison, a spot weld of an inflated 316L panel sample, filled voluntary with a caustic solution of pH 11.5 (25 °C), failed after 90 h of testing (intergranular cracking at the spot weld), while an identical sample containing AVT water of pH 9.0 (25 °C) survived without damage.
The results of these tests, performed during 1986 and 1997, have never been published and therefore are presented more in detail in this paper since corrosion in hydraulic circuits is also an issue of ITER.
Up to day, the TS cooling water plant operates with an above mentioned water treatment and no water leaks have been detected on in-vessel components originating from water corrosion at high temperature and high pressure.

IMPACT INTERACTION BETWEEN MODULE AND VACUUM VESSEL. DYNAMIC TEST PROGRAM AND ANALYSIS

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The blanket modules of ITER are mechanically attached to the vacuum vessel with four radial flexible supports and three prismatic keys. The keys are built on the vacuum vessel and react to the radial torque, the poloidal and toroidal forces, and let the module to expand. They are fitted with bronze pads sliding against the key-ways of the modules during relative expansion. The electrical insulation is made by ceramic coating on the hidden face of the pad. The pads are sized for an average compressive stress of 100 MPa, to exclude the yielding of the bronze, the only condition that may damage the ceramic coat.

The blanket assembly on the vacuum vessel is achieved owing to clearances betwen the module and the keys, wich compensate the manufacturing tolerances. The gap remaining after the assembly allows some acceleration of the module before reaching the contact with the key and produces a dynamic amplification of the reaction force under the electromagnetic loads. There is now ample analytical evidence pointing to essentiale exceeding of the yielding stress inasmuch as the impact spot usially is on the edge of the bronze pad.

The dynamic amplification depends on the many factors, the mains are: initial clearence between module and key, magnitude and duration of loading, mass and stiffness of the module/vessel structure, temperature distribution and energy dissipation. So, the dynamic analysis of the blanket should provide for all these factors. However, some of them couldn't be valued analytically but experimentally only. For example, energy dissipation (damping), wich determines the number of the high frequency overstresses in the contact area, while the module is pressed to the key. Another issue is a possible rising of the gap between module and key as a result of residual plastic deformation, which leads to impact increment under the next plasma disruptions.

The paper will present the program of the dynamic test of the different keys and bronze pads covered with the ceramic insulation. Also, the design of the test facility, analytical grounds the experimental conditions meet the facts, the methods for experimental determination of the damping, and the pilot analysis of the expected experimental results will be discussed.

COMPUTATIONAL THERMO-FLUID EXPLORATORY DESIGN ANALYSIS FOR COMPLEX ITER FW/SHIELD COMPONENTS

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Engineers in the ITER US Party Team used several computational fluid dynamics codes to evaluate design concepts for the ITER first wall panels and the neutron shield modules. The CFdesign code enabled them to perform design studies of modules 7 and 13 very efficiently. CFdesign provides a direct interface to the CAD program, CATIA v. 5. The geometry input and meshing are greatly simplified. CFdesign is a finite elements code, rather than a finite volume code. Flow experiments and finite volume calculations from SC-Tetra, Fluent and CFD2000 verified the CFdesign results. Several new enhancements allow CFdesign to export temperatures, pressures and convective heat transfer coefficients to other finite element models for further analysis. For example, these loads and boundary conditions directly feed into codes such as ABAQUS to perform stress analysis. In this article, we review the use of 2 and 4-mm flow driver gaps in the shield modules and the use of 1-mm gaps along the tee-vane in the front water header to obtain a good flow distribution in both the first wall and shield modules for 7 and 13. Plasma heat flux as well as neutron heating derived from MCNP calculations are included in the first wall and shield module analyses. We reveal the non-uniformity of the convective heat transfer coefficient inside complex 3-d geometries exposed to a one-sided heat flux and non-uniform volumetric heating. Most models consisted of 3 to 4 million tetrahedron elements. We obtained temperature and velocity distributions, as well as pressure drop information, for models of nearly exact geometry compared to the CATIA fabrication models. We also describe the coupling to thermal stress analysis in ABAQUS. The results presented provide confidence that the preliminary design of these plasma facing components will meet ITER requirements.

*Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under Contract DE-AC04- 94AL85000.

CONCEPTUAL DESIGN ON STRUCTURE AND COOLING CHANNEL OF ITER UPPER PORT PLUG

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This study has performed conceptual design on structure and cooling channel for the upper port plug of the International Thermonuclear Experimental Reactor (ITER), in which electron cyclotron heating (ECH) launcher and various diagnostic modules will be installed with the same structure.
There are twelve diagnostic plugs and four ECH plugs at the upper port in ITER Tokamak. The use of the same port plug structure is beneficial for installation of diagnostic modules and ECH launcher from the viewpoint of cost reduction and simple RH maintenance. The diagnostic modules have rectangular cross-section and ECH modules have trapezoidal cross-section with the lower part wider. Here was suggested the bolt-jointed common structure of inverted-U shape beam and bottom plate, where the diagnostic and ECH modules are installed onto the bottom plate and then the assembly is bolted to the inverted-U beam from the bottom. The common structure of Inverted-U type was evaluated by considering several aspects, such as installation, remote handling (RH) maintenance, cooling line connection, manufacturing, and structural stiffness.
For the inverted-U port plug structure developed here, this paper proposed a network of water channel for cooling and baking. Pressurized water as working fluid has to be supplied into the whole port plug. It consists of the structure, diagnostic/shielding modules fixed onto the bottom plate, and the blanket shield module (BSM) attached to the front. The internal water ways for these three components were designed in the direction that would not only minimize the RH connections, flow restrictors, and the length of water-vacuum welding, but also make the welding reliable. Independent coolant loops were composed for three parts of the structure, BSM, and diagnostic/shielding modules with bottom plate. These loops, therefore, make it possible to perform the leakage test for each one separately.
Finally hydraulic analysis has been performed with ANSYS in order to decide proper size and number of cooling channels by checking the flow balance and overall pressure drop. Temperature distribution was also evaluated by thermal-hydraulic analysis with CFX code.

PS1-1107

STEADY STATE AND TRANSIENT THERMAL-HYDRAULIC CHARACTERIZATION OF FULL-SCALE ITER DIVERTOR PLASMA FACING COMPONENTS

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In the frame of the activities related to ITER divertor R&D, ENEA CR Brasimone was charged by EFDA (European Fusion Design Agreement) to investigate the thermal-hydraulic behaviour of the full-scale divertor plasma facing components, i.e. Inner Vertical Target, Dome Liner and Outer Vertical Target, both in steady state and during draining and drying transient.
More in detail, for each PFC, the first phase of the work is the steady state hydraulic characterization which consists of:
- measurements of pressure drops at different temperatures;
- determination of the velocity distribution in the internal channels;
- check the possible insurgence of cavitation.

The subsequent phase of the thermal-hydraulic characterization foresees a testing campaign of draining and drying procedure by means of a suitable gas flow. The objective of this experimental procedure is to eliminate in the most efficient way the residual amount of water after gravity discharge. In order to accomplish this experimental campaign a significant modification of CEF1 loop has been designed and realized.
This paper presents, first of all, the experimental set-up, the agreed test matrix and the achieved results for both steady state and transient tests. Moreover, the level of the implementation of a predictive hydraulic model, based on RELAP 5 code, as well as its results are described, discussed and compared with the experimental ones.

PS1-1108

AN OVERVIEW OF THE US WORK TO COMPLETE THE DESIGN OF BLANKET SHIELD MODULES 7, 12 AND 13 FOR THE ITER PROJECT

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Part of the US contribution to the ITER project is Blanket Shield Modules (BSM) number 7, 12 and 13 comprising about 20% of the ITER First Wall. These modules are all in the upper half of the vacuum vessel. Starting from outlines of each style of module, general design rules contained in the ITER Design Description Document (DDD), design memos, and CATIA interface drawings, the US Team has worked to fill in the details of the internal water cooling passages, slits to control eddy currents, and methods for efficient and reliable manufacturing of the BSM. Our analysis begins with nuclear heating assessment of complex 3D structures containing water, copper, and steel carried out by the University of Wisconsin using a version of Monte Carlo N-Particle Transport Code (MCNP) that connects directly to CATIA to get the geometry. Computational Fluid Dynamics (CFD) analysis of the coolant flow distribution and pressure drop in a shield module provides the basis for thermal transfer from the BSM to the coolant. The size and position of coolant passages are adjusted to optimize the heat transfer and eliminate hot spots. ITER specified major disruption (MD) and downward vertical disruption (VDE) events are used to calculate the currents induced in the BSM. In this modeling it is necessary to include the vacuum vessel and other BSM near the modules of interest. In order to benchmark the OPERA Electromagnetic Code against the one used for the DDD analysis, we calculated eddy currents and forces on all 18 BSM in a simplified model that matched analysis by Japan. The eddy current forces are used to determine the torque and net force on the BSM. These forces are compared to the load capacity of the mounts and adjustments made to eddy current control slits as needed. Dynamic analysis of the eddy current induced stresses on the BSM and mounts are performed using the ABAQUS code. Static thermal and pressure stresses are calculated using the temperature distributions from CFD analysis. The primary and secondary stresses are compared to the allowables specified in the ITER Structural Design Criteria to determine the suitability of the design to the ITER needs. Manufacturing processes are being created through a series of mockups and prototypes of sub-scale parts. This paper will describe the results of these analyses that have led to a Preliminary Design for the US contribution of BSM.

* Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy’s National Nuclear Security Administration under contract DE-AC04-94AL85000.

ELECTROMAGNETIC ANALYSIS OF TRANSIENT DISRUPTION FORCES ON THE ITER SHIELD MODULES

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There are potential abnormal operating environments where the disruption of the plasma currents inside a tokamak induce eddy currents in the shield modules. These currents interact with the large magnetic fields to produce forces in the modules which could potentially cause mechanical failure in the modules and vacuum vessel. For this reason the design and qualification of the ITER shield modules requires appropriate high-fidelity electromagnetic simulations that capture the physics of these situations. These simulations need to include an accurate representation of the disruption currents as well as an accurate electromagnetic model of the shield modules. The purpose of this presentation is to describe the electromagnetic analysis that has been completed using the OPERA-3D product to characterize the forces on the shield modules allocated to the US. We first describe the electromagnetic model of the system which consists of the disruption currents and the shield modules attached to the vacuum vessel. The disruption currents are represented in OPERA-3D using superposition of a large number of solenoids with independent time variation to account for the spatial and temporal variation of the plasma current and position. In addition, the simplified electromagnetic model of the shield modules will be described and discussed. Once the modeling has been described the simulation results will be presented. The force computation will also be presented and the results discussed. These forces are then used by a mechanical analysis program to compute stresses and torques on a module during the disruption of the plasma currents.

Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

PS1-1110

DESIGN OF THE ITER TOKAMAK ASSEMBLY TOOLS

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ITER (International Thermonuclear Experimental Reactor) Procurement allocation among the seven Parties, EU, JA, CN, IN , KO, RF and US have been decided in Dec. 2005. ITER Tokamak assembly tools is one of the nine components allocated to Korea for the construction of the ITER. Assembly tools except measurement and common tools are supplied to assemble the ITER Tokamak and classified into 9 groups according to components to be assembled.

Among the 9 groups of assembly tools, large-sized Sector Sub-assembly Tools and Sector Assembly Tools are used at the first stage of ITER Tokamak construction and need to be designed earlier than seven other assembly tools. ITER proposed Korea to accomplish ITA (ITER Task Agreement) on detailed design, manufacturing feasibility and engineering design specification of large-sized tools such as Sector Sub-assembly Tools and Sector Assembly Tools in Jan. 2007. Based on the concept design by ITER, Korea carries out ITA on detailed design of large-sized Sector Sub-assembly and Sector Assembly Tools until Sep. 2007.

The Sector Sub-assembly Tools mainly consist of the Upending, Lifting, Vacuum Vessel Support and Bracing, and Sector Sub-assembly Tool. The Sector Assembly Tools mainly consist of the Toroidal Field (TF) Gravity Support Assembly, Sector In-pit Assembly, TF Coil Assembly, Vacuum Vessel (VV) Welding Tool. The design of Sector Sub-assembly Tools and Sector Assembly Tools are described herein.

PS1-1111

A STUDY ON THE THERMAL HYDRAULIC AND THERMAL ANALYSES OF THE ITER THERMAL SHIELD

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The detailed design of ITER thermal shield is progressed by Korea ITER project team, based on the design done by ITER international organization and other participant teams. The thermal shield system of ITER minimizes heat loads transferred by thermal radiation and conduction from warm components to the components and structures that operate at 4.5 K. Detailed analysis of the thermal loads is very important for ensuring that the thermal shields can handle the loads within the specified boundary conditions. This paper presents some results of thermal-hydraulic and thermal analyses for the plasma operation state (POS) and the vacuum vessel baking state (BOS) to verify the design of the thermal shield. We calculated the radiation heat loads of the system. And inlet and outlet helium temperature, mass flow rate and pressure drop at the cooling pipes of the thermal shield are also calculated. The results of the analyses can be adopted in the detailed design of ITER thermal shield.

DESIGN PROGRESS OF THE VV SECTORS AND PORTS TOWARDS THE ITER CONSTRUCTION

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The ITER vacuum vessel (VV) is an all-welded torus-shaped double-wall structure with stiffening ribs between the shells. The VV main function is to provide the high-vacuum and primary confinement boundary. The vessel also supports in-vessel components such as the blanket modules and the divertor cassettes. Along with these components, the VV provides radiation shielding – the neutron heat is removed by water circulating between the shells. To satisfy the manufacture and assembly needs, the VV consists of nine sectors. To provide access inside the vessel for auxiliary plasma heating, diagnostics, vacuum pumping and other needs, the VV is equipped with upper, equatorial and lower ports. The upper and regular equatorial ports are occupied with the port plugs. In addition, there are three ports at the equatorial level dedicated for neutral beam (NB) injection.

As the ITER construction phase approaches, the VV design has been improved and developed in more detail with the focus on improved manufacture and reduced cost. Based on achievements of manufacturing studies being performed in cooperation with industry, design improvement of the typical VV sector (#1) has been nearly finalized. Design improvement of other sectors is in progress - in particular, of the VV sectors #2 and #3 which interface with the NB ports. For all sectors, the concept for the in-wall shielding has been improved and developed in more detail.

The design progress of VV sectors #2-3 has been accompanied by progress in the NB port design (including the beam-facing components to handle the heat flux input of the neutral beams). Design of other port structures has also progressed. Thus, supporting and sealing components between the port plugs and the ports have been further developed with the focus on improved structural performance and maintenance. At the lower level, there are full-size ports, and the pipe feedthroughs and local small penetrations. Design of all port structures at this level has progressed towards completion.

At this stage of the project, special attention is paid to the code related aspects of the machine. For the vacuum vessel, an addendum to the existing nuclear code is being developed to address the VV design/manufacture peculiarities and facilitate its acceptance by the licensing authorities. This work is being performed by the EU organizations in close cooperation with the ITER design team.

Details of the current VV design and results of the related studies are reported in this paper.

PS1-1113

RECENT PROGRESS OF ITER VACUUM VESSEL RELATED DESIGN ACTIVITIES IN KOREA

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Based on the design done by ITER international organization and other participant teams, the more detail engineering design of ITER vacuum vessel port and support is progressed by Korea ITER project team. In this work, the recent design elaboration and manufacturing feasibility study from 2006 mainly focused on the equatorial and lower port are reported. The fabrication status and test plan of vacuum vessel support mock-up are also introduced. In addition, refined thermal hydraulic analysis of vacuum vessel in-wall shield region is presented.

A PROPOSAL OF ITER VACUUM VESSEL FABRICATION SPECIFICATION AND RESULTS OF THE FULL-SCALE PARTIAL MOCK-UP TEST

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The structure and fabrication methods of the ITER vacuum vessel have been investigated and defined by the ITER international team. However, some of the current specifications are very difficult to be achieved from the manufacturing point of view and will lead to cost increase.
In the mock-up fabrication, it is planned to conduct the following items:
1. Feasibility of the Japanese proposed VV structure and fabrication methods and the applicability to the ITER are to be confirmed;
2. Assembly procedure and inspection procedure are to be confirmed;
3. Manufacturing tolerances are to be assessed;
4. Manufacturing schedule is to be assessed.
This report summarizes the Japanese proposed specification of the VV mock-up describing differences between the ITER supplied design. General scope of the mock-up fabrication and the detailed dimensions are also shown.
In the VV fabrication, several types of weld joint configuration will be used. This report shows the joint configurations proposed by Japan to be used for the inner shell connection, the rib-to-shell connection and outer shell connection, and the housing-to-shell connection, respectively. Non-destructive testing considered to be applied to each joint configuration is also presented.
A series of the fabrication and assembly procedures for the mock-up are presented in this report, together with candidates of welding configurations.
Finally, the report summarizes the results of mock-up fabrication, including results of non-destructive examination of weld lines, obtained welding deformation and issues revealed from the fabrication experience.

DESIGN ANALYSIS OF THE ITER DIVERTOR

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The divertor is one of the most challenging components of the ITER machine. Its function is to reduce the impurity in the plasma and consists essentially of two parts: the plasma facing components (PFCs) and a massive support structure called the cassette body (CB).
Considerable R&D effort (developed by EFDA CSU GARCHING and the ITER International Team together with the EU Associations and the EU Industries) has been spent in designing divertor components capable of withstanding the expected electromagnetic (EM) loads and to take into account the latest ITER design conditions.
In support of such efforts extensive and very detailed Neutronic, Thermal, EM and Structural analyses have been performed.
A summary of the analyses performed will be presented. One of the main result is a typical exercise of integration between the different kind of analyses and the importance of keeping the consistency between the different assumptions and simplifications.
The models used for the numerical analyses include a detailed geometrical description of the CB, the inlet, outlet hydraulic manifolds, the CB to vacuum vessel locking system and three configurations of the PFU. The effect of electrical bridging, both in poloidal and toroidal direction, of the PFU castellation, due to a possible melting at the W mono-block or tiles, occurring during the plasma disruptions, has been analyzed.
For all these configurations 2 VDE scenarios including the effect of the Toroidal Field Variation and the HaloCurrent with the related out of plane induced EM forces have been extensively analyzed and a detailed poloidal and radial distribution of the nuclear heating has been used for the neutronic flux on the divertor components.
The aim of this activity is to produce a comprehensive design and assessment of the ITER divertor via:
• The estimation of the neutronic heat deposition and shielding capability;
• The calculation of the related thermal and mechanical effects and the comparison of the computed stress with the design criteria for Category 1 loads;
• The estimation of EM loads due to the off-normal events and calculation of the related mechanical stress; the computed stress are then compared with the design criteria for Category 2 and 3 loads.

DETAILED ELECTROMAGNETIC NUMERICAL EVALUATION OF EDDY CURRENTS INDUCED BY TOROIDAL AND POLOIDAL MAGNETIC FIELD VARIATION AND HALO CURRENTS

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Considerable R&D effort has been spent in designing Divertor components capable of withstanding the expected electromagnetic (EM) loads and to take into account the latest ITER design conditions.

The EM loads are one of the most critical load conditions for the Divertor during Plasma Disruptions and their detailed evaluation is mandatory for the correct dimensioning of the Divertor.

The EM loads during plasma disruptions can by considered produced by three main causes:

1- the TFV (Toroidal Field Variation) during the TQ (Thermal Quench) and CQ (Current Quench);

2- the HC (Halo Currents);

3- and the PFV (Poloidal Field Variation) during the fast CQ phase.

The TFV due to TQ (responsible of a traction force toward the plasma), is a very fast phenomenon, and the maximum of these eddy currents and the related resultant load could be analytically estimated assuming toroidal field flux conservation. The eddy currents induced via TFV during the CQ is a much slower phenomenon and could be evaluated only by numerical analyses.

For the past, the loads due to TFV and HC were estimated analytically on the basis of some simplifying assumptions, and almost all the EM efforts were dedicated to analyze the eddy currents due to the large PFV.

In the present analyses the evaluation of the EM loads due to PFV has been repeated taking into account the new ITER reference disruption events and the most recent divertor design; furthermore to achieve a more complete knowledge of all the EM loads acting on the Divertor components, with particular attention to the PFC multilink connections, more detailed analyses of eddy currents induced by TFV and HC have been performed.

Indeed the knowledge of a more correct sharing of these currents among the different Divertor components gives useful information not only on the resultant loads but even on the local loads; the importance of the knowledge of the current distribution inside the divertor is also related to the fact that the poloidal currents, via interaction with poloidal magnetic field strongly dependent on position, are responsible of out-of-plane forces.

While the effects of PFV have been analyzed using the EM-zooming procedure used for the past Divertor analyses, new numerical approaches for the evaluation of the eddy currents due to TFV and HC have been developed. It has been possible to maintain the same mesh of the divertor while the surrounding mesh was changed to match the requirements of the three different problems, that require different boundary conditions and kind of excitations: orthogonal field at the boundary and poloidal excitation currents for TFV and the HC case, tangent field and toroidal excitation currents for the PFV case. Furthermore the PFV and the TFV problems are at “imposed induced voltage”, while the HC problem is at “imposed current”.

PROGRESS IN NEUTRONICS FOR THE ITER ECRH LAUNCHER

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This paper presents the essentials of neutronics analyses for the ITER upper port Electron Cyclotron Resonance Heating (ECRH) launcher design development. The launcher serves for the injection of the mm-waves into the plasma for its stabilization and current drive. The mm-waves enter into the launcher from its back side through the diamond windows served for a vacuum and tritium barrier, then pass along the waveguide (WG) channels, and are finally focused precisely on the plasma magnetic surfaces. The accuracy of mm-waves positioning is achieved by the system of focusing and steering mirrors. In the recent Front Steering (FS) launcher design the steering mirrors are located near the plasma first wall. Therefore they are exposed to high neutron loads. This makes detail neutronics modeling inevitable. A distinguishing feature of the launcher design is a radiation shield of several meters thick inside of which long and narrow (6 cm in diameter) eight WG channels are arranged. The neutronics analyses have been performed with the Monte Carlo code MCNP considering 3D heterogeneous geometry of the ITER machine and a detail description of the critical parts of the launcher. The CAD to MCNP interface program McCad has been used to generate the launcher 3D MCNP models from the CAD design data.
Nuclear responses inside of the launcher and adjacent ITER components have been evaluated to satisfy the nuclear design limits. Among them is the fast neutron fluence on the diamond windows at the distance of 5 meters from the first wall. Proper arrangement and material compositions of the shield blocks in the launcher structure has been obtained which guarantee for design limits. Radiation damage of materials, the nuclear heating, and radiation loads on critical components of the launcher, superconductive magnets, and vacuum vessel have been estimated.
An elaborated use of variance reduction techniques has been made in MCNP transport calculations. The trajectories for two types of neutrons could be distinguished in a deep-penetration transport in the launcher: 1) neutrons attenuated in the bulk shield, and 2) neutrons streamed in the void WG channels. The energy and angular distributions of these neutrons are different. The point detector technique of MCNP was applied for calculation of the streaming neutrons in the WG channels. Particles splitting and Russian roulette are used for the neutron transport in the bulk shield. Particle weights are substantially decreased along the splitting in such high-biased calculations, reducing calculation time without distortion of results. If it is necessary to include the contribution from the WG streaming neutrons, then the particles are split with more flat variation of the neutron weights, MCNP tallies are resized, and longer calculation time is consumed.
Shutdown dose rate calculations have been performed for the assessment of personnel access for maintenance at the launcher back side. A decay gamma transport has been taken into account in such calculations by means of Rigorous 2-Step (R2S) and Direct 1-Step (D1S) methods. In R2S method the radiation MCNP transport and FISPACT activation calculations are performed in sequential steps, while in D1S the neutron transport and decay gammas calculations are merged into the one MCNP run.

ITER BLANKET MODULE #17 SHIELD BLOCK DESIGN AND ANALYSIS

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The shield block reference design of the typical ITER blanket module has a number of grave disadvantages, precarious with relation to nuclear safety of the reactor. The main problems may arise when innage of the parallel cooling passages both in the first wall and in the shield block. Vapor locking in a radial channel with flow insert driver is very probable. Another problem, as a result of the same reason, is draining and dehydration of the coolant system. Then the highly dense packing of the radial channels in the collector array brings an essential flow irregularity. Customary as a rule, the lack of coolant is observed in the last channels, nearest to the outside, most heated surface of the shield block. A local boiling is possible in these dead spaces of coolant system. In consequence of the radial flow irregularity the cooling in the upper box header, directly under the first wall, may be extremely poor. Among the other imperfections one should note the large frontal figured lids, which overburden at welding and give to rise of stresses and shrinkages, and as a result, the large share of irreparable spoilage.

The paper represents an alternative design of the shield block coolant system with predominantly sequential flow circuit. The cooling channels are drilled from the frontal side as inclined transverse holes. The open drilling ends are combined in pairs with milled grooves and welded with small lids. This gain the following advantages: the lids may have smaller thickness (7 mm instead 20 mm), the cooling passengers are placed closer to the lateral and upper sides and make cooling better, the welding stress and shrinkages are reduced, there are no any dead spaces of coolant, and the water fillup and draining are substantially improved.

The listed hydraulic and thermo mechanical problems have been analysed with help of 3D models in ANSYS CFX program. The models include both the cooling space filled by water and the solid part of shield block. Thus the conjugate hydraulic and thermo structural analysis is solved simultaneously with account of real geometry and local flow turbulence. The results of the analysis are assumed as a basis of design modification to eliminate the existent disadvantages of the reference design.

CURRENT STATUS ON DETAIL DESIGN AND FABRICATION TECHNIQUES DEVELOPMENT OF ITER BLANKET SHIELD BLOCK IN KOREA

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The allocation of components and systems to be delivered to ITER on an in-kind basis, was agreed between the ITER Parties. Among parties, Korea agreed to procure inboard blanket modules #1, #2 and #6, which consists of FW and shield block. Regarding shield block the detail design and Fabrication techniques development have been undertaken in Korea. Especially manufacturing feasibility study on shield block had been performed and some technical issues for the fabrication were selected. Based on these results, fabrication techniques using EB welding are being developed. Meanwhile, the detail design of inboard standard module has been carried out. The optimization of flow driver design to improve the cooling performance was executed. And, thermo-hydraulic analysis on half block of inboard standard module was performed. In this study, current status and some results from Fabrication techniques development on ITER blanket shield block are described. The detail design activity and results on shield block are also introduced herein.

ASSESSMENT OF A WATER HYDRAULICS JOINT FOR RH OPERATIONS IN THE DIVERTOR REGION

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Due to the high level of radiations, all the nominal maintenance in the divertor region of ITER will be carried out with help of robotic means. In reduced volumes, hydraulic applications can provide powerful actuators. They become an interesting technology to build a heavy duty manipulator for operations in space constrained areas. Oil hydraulics can not ensure the cleanliness level required for all maintenance operations in the vacuum vessel. Therefore, pure water hydraulics proposes a good alternative to oil and developments are today focusing on that direction. Although basic hydraulic elements like pumps, on-off valves, filters running with pure water are already available on the market, actuators are not so many and generally limited to linear motions. Fine control of the joint is achieved with help of servovalves. Today's off the shelf products are only adaptations from standard oil servovalves and are not specifically designed for water use. Operational experience for these products shows short lifetime expectancy and could not last a complete shutdown.

Starting from the oil hydraulic version CEA with help of Cybernetix redesigned for water applications the elbow vane actuator of a Maestro arm, a six-degrees-of-freedom hydraulic manipulator used in decommissioning activities.

In parallel with help of In-LHC, CEA developed a servovalve for water hydraulic applications that fits the space constraints of a Maestro manipulator. This prototype is a pressure-control valve. To a current input this servovalve supplies a very accurate pressure difference output instead of a flow rate in the case of flow control servovalve that are generally used in that kind of applications. The advantage is the improvement of the performances and stability of the force control loop.

This paper presents the performances of the modified vane actuator and its servovalve.

Both static and dynamic responses of the servovalve prototype with and without actuator are presented. Position and force control loops were assessed and endurance tests were performed. Loads and trajectories applied on the mock-up during the trials were defined according to representative Remote Handling tasks during maintenance operations.

PS1-1202

PROGRESS OF R&D AND DESIGN OF BLANKET REMOTE HANDLING EQUIPMENT FOR ITER

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ITER blanket remote maintenance is carried out in the vacuum vessel (VV) by the vehicle type remote handling equipment. The number of the blanket modules is about 400 and installed in the VV. The dose rate of gamma ray is expected about 500 Gy/h during blanket maintenance. The visual information in the VV for maintenance will be limited even though the cameras are installed in the VV. Application of the simulation system, which is synchronized with actual remote handling equipment, is therefore useful for the reliable maintenance operation. The system, which simulate the position of the equipment based on the respective position data of the motors obtained from the existing control system through LAN, was newly developed using a 3D simulation software, ENVISION. The simulation system can detect the interference such as collision between handling module by equipment and adjacent modules during module installation. The simulation system also can provide additional viewing information such as virtual removal or change of the transparency of the obstacles in the blind angle during module installation.

Development of dry lubricant is also a key issue for remote handling equipment to prevent the lubricant oil from spreading in the VV as much as possible. Diamond-like carbon (DLC) coating is a candidate and was applied to the performance tests for the feasibility of dry lubricant. The "pin on disk" tests were adopted to examine the basic performances. The test results have satisfied the requirement as follows: the contact pressures and life cycles are 2.5 GPa up to 30000 cycles, and 4.2 GPa up to 10000 cycles to the requirement of 2 GPa up to10000 cycles, respectively.

In addition to the R&D, the design of remote handling equipment has been updated according to the design changes such as blanket segmentation and structure, taking account of the interface between modules and remote handling equipment. The stress and kinematic analyses were performed for the design of the vehicle manipulator and rail in order to avoid the interference between modules and vehicle manipulator. The major outputs are the adoption of the double helical gears instead of spur gears and guide roller mechanism combined with support pad mechanism instead of separate mechanisms for the optimization of in the vehicle structure.

PS1-1203

REMOTE HANDLING DYNAMICAL MODELLING: ASSESSMENT ON NEW APPROACH TO ENHANCE POSITIONING ACCURACY WITH HEAVY LOAD MANIPULATION

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In vessel maintenance work in Fusion Tokamak will be carried out with help several sets of robotic devices. Heavy loads handling in constrained space is identified by all players of the RH community as a key-issue in the latest Fusion Tokamak facilities. To deal with high-level dexterity tasks, high payload to mass ratio and limited operating space, RH equipment designers can only propose systems whose mechanical flexibility is no longer negligible and need to be taken into account in the control scheme. Traditional approaches where control system only includes a linear model of deformation of the structure leads to poor positioning accuracy. Uncontrolled or under evaluated errors could be damaging for in-vessel components during maintenance operations in the Tokamak facility. To address the control of complex flexible systems, we will investigate the use of specific mechanical software that combines both finite element and kinematical joints analyses, with a strong-coupled formulation, to perform system dynamics simulations. This procedure will be applied on a single axis mock up robotic joint with highly flexible structure. A comparison of experimental results with the traditional linear approach and the specified software model will be carried out. Benefits introduced by this new approach will finally be assessed in view of RH design or specification in the field of RH in Fusion Tokamak scale such as ITER.

HIGH HEAT FLUX TEST WITH THE HIP BONDED MOCK-UPS FOR THE ITER FIRST WALL

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The first wall (FW) of the International Thermonuclear Experimental Reactor (ITER) is an important component which directly faces with the plasma and therefore, it is subjected to a high heat and high neutron loads. The FW is composed of a beryllium (Be) layer as a plasma facing material, a copper alloy (CuCrZr) layer as a heat sink and type 316L authentic stainless steel (SS316L) as a structure material. To fabricate the FW, the Hot Isostatic Pressing (HIP) bonding method has been investigated. To investigate the thermo-mechanical performance of the FW, including the integrity of the HIP bonded interfaces, high heat flux (HHF) tests are essential. In this paper, the results of the HHF test for Cu/SS performed in JEBIS (JAEA electron beam irradiation stand) and the preparation process of the HHF test for a Be/Cu mock-up in TSEFEY-M facility (Russia) are introduced;

The optimum joining condition of a HIP for the ITER FW has been developed by using Be of a S-65C grade, CuCrZr, and SS316L. Here, CuCrZr/SS316L (tube and block) and Be/CuCrZr including SS316L tube mock-ups were fabricated to investigate their integrity for joining parts through several tests. They were successfully HIPped at 550 oC, 150 MPa, and 1 hour for Be/CuCrZr and at 1050 oC, 100 to 150 MPa, and 2 hours for CuCrZr/SS316L. In order to be installed in JEBIS, thermocouples and manifolds are added to the Cu/SS mock-up. Dimensions of the mock-up are 101 mm long, 50 mm wide and 52 mm thick with two circular cooling tubes (8mmID). Five thermocouples are installed to measure the temperature in a mock-up according to a certain distance from a heat source. Two manifolds and connecting pipes were prepared for the coolant. For the HHF test in the TSEFEY-M facility, Be/Cu mock-ups were fabricated. Dimensions of the mock-ups are 50 mm long, 50 mm wide, and 32 mm thick (10 mm of Be tile and 22 mm of Cu alloy). Two circular tubes (10mm ID) were inserted for a cooling.

JEBIS was used as a high heat flux test facility for the Cu/SS mock-ups since there is no high heat flux test facility in Korea. For the preliminary analysis with ANSYS-10 to establish the test conditions according to the water cooling system and e-beam capacity in the JEBIS, a test was performed with 5 MW/m2 of a heat flux, 7 m/sec of a cooling water speed (0.1 MPa, 25 oC), and a 45 sec duration (15 sec heating and 30 sec cooling). Temperature responses during the first and the 1000th cycle agreed very well and they also agreed well with the analysis result. However, after the 1000th cycle, the temperature became higher than expected and the test was stopped. Delaminations were found in the Cu/SS mock-up. From the constant strain fatigue curve for SS316L and CuCrZr, the expected life times are 2310 and 780 cycles, respectively.

HHF test for the Be/Cu mock-ups are being prepared and will be performed with the TSEFEY-M facility in Russia. The test conditions were established from an analysis with ANSYS-10 in the same way as the Cu/SS mock-up; the heat flux was assumed to be 3.2 MW/m2 so as not to exceed the Be temperature limitation (630 oC); water cooling conditions was determined from the conditions (25 oC and 2 MPa). For a sufficient cooling, the water speed and heat transfer coefficient in the tubes were assumed to be 10 m/sec and 3162 5W/m2K, respectively.

TEMPERATURE DEPENDENCE OF BLISTERING AND DEUTERIUM RETENTION IN TUNGSTEN

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Tungsten has been selected as the plasma facing material (PFM) at the divertor baffles and dome in the ITER design, because tungsten has a very high threshold energy for sputtering, a high melting point and a very low tritium inventory. However, blistering can occur at tungsten surface, even if the ion energy is too low to create displacement damage such as vacancies. Tungsten blistering could lead to instability of the plasma due to high-Z impurity release into the core plasma and sudden gas recycling. In addition, an increasing tritium inventory in the near surface region of PFCs could become a significant safety issue during the exchange process of the PFCs. Therefore, blistering and deuterium retention in tungsten exposed to high fluences (up to 1E27 D/m2) of high flux (1E22 D+/m2/s) and low energy (38 eV) deuterium plasma were examined in the temperature range of 315 K to 1000 K with scanning electron microscopy (SEM), focused ion beam (FIB) and thermal desorption spectroscopy (TDS).

At 315 K, only sparse low-dome blisters with a chord of a few microns or less appeared even the fluence was increased to 1E27 D/m2. At around 400 K, the blisters became much denser and the dome of blisters became a little higher. Peculiar change occurred around 500 K, where two kinds of blisters appeared. One is the large blisters with sizes of a few tens of microns and varying ratios of height against chord (up to 0.6), and the other is the small blisters with chords of less than a few microns and large ratio of height against chord (about 0.7). In high temperature region (higher than 600 K), the blisters became much sparser with the increasing temperature and disappeared at 1000 K. In addition, the phenomenon of blister bursting with a tail, or partially-opened or fully-opened lid was found on some grains after plasma exposure or TDS experiments.

During TDS experiments, bursting release with sadden peaks was observed, suggesting the bursts of blisters. Deuterium retention showed the maximum around 500 K, corresponding to the appearance of two kinds of high-dome blisters. Furthermore, the amount of deuterium retained in tungsten increased with the increasing fluence, roughly following the proportional relationship with the root of the exposure time. This implies that deuterium diffusion could play an important role in retention.

AN OVERVIEW OF FUEL RETENTION AND MORPHOLOGY IN A CASTELLATED TUNGSTEN LIMITER

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All plasma-facing components (PFC) in ITER will be castellated, i.e. composed of small blocks separated by narrow grooves (~ 0.5 mm) in order to reduce thermally-induced stress. Narrow grooves of castellation are considered as a potential trap for species migrating to these gaps. Therefore, co-deposition of fuel together with material eroded from the wall may be decisive for the overall tritium inventory in a device with a huge number (over one million in ITER) of grooves. This calls for detailed studies of castellated structures exposed to the plasma for long- [1] or short-term [2] in present-day tokamaks.

A castellated tungsten test limiter (macro-brush structure) was exposed to plasma discharges in the TEXTOR tokamak operated with graphite main limiters and an Inconel liner [3]. The limiter was composed of twelve detachable segments, each brased to a copper base. Dismantling of the segments enabled the analysis of surfaces inside the castellation. The emphasis was on the determination of: (i) deposition and fuel retention on the plasma-facing surface and in the gaps and (ii) material mixing and new compound formation inside the castellated structure. The study performed by means of nuclear reaction analysis (NRA), Rutherford backscattering spectroscopy (RBS), electron microscopy, X-ray diffraction (XRD) and other methods has brought several essential results.

(a) Deuterium retention on plasma-facing surfaces and in the castellation of metal PFC is strongly related to the co-deposition with carbon;

(b) Both carbon and deuterium are detected only in narrow belt, a few mm broad, down the gap with the decay length of about 1 -1.5 mm;

(c) The presence of copper droplets and tungsten oxide has been identified in the gaps of the macro-brush limiter. This is probably the first-ever identification of tungsten oxide on PFC. Different pathways leading to the oxide formation are considered.

The implications of these results for a long pulse operation of a tokamak with castellated wall components are discussed.

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PS1-1308

SIMULATION OF DUST PRODUCTION IN ITER TRANSIENT EVENTS

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The tritium retention problem is a critical issue for the tokamak ITER performance. Tritium is trapped in redeposited T-C layers and at the surface of carbon dust, where it is retained in form of various hydrocarbons. The area of dust surface and hence, the amount of tritium deposited on the surface depends on the dust amount and of the dust sizes. The carbon dust appears as a result of brittle destruction at the surface of the carbon fibre composite (CFC) which is now the reference armour material for the most loaded part of tokamak divertor.

Stationary heat flux on the ITER divertor armour does not cause its brittle destruction and does not produce dust. However, according to the modern understanding of tokamak fusion devices performance, the most attractive regime of ITER operation is the ELMy H mode. This regime is associated with a repetitive short time increase of heat flux at the CFC divertor armour of 2-3 orders of magnitude over its stationary value during edge localized modes (ELMs). Under influence of these severe heat shocks CFC armour can crack due to the thermostress, producing a dust of carbon. Besides, a carbon dust produced during disruptions due to brittle destruction of the armour under influence of thermoshock.

Most of the modern tokamaks do not produce the ELMs powerful enough to cause CFC brittle destruction at the divertor surface, except of very special regimes in JET. This is why the CFC erosion and dust production could be investigated now only theoretically and experimentally in plasma guns and electron beam facilities.

Simulation of the CFC brittle destruction has been done using the code PEGASUS already developed and tested in FZK for simulation of erosion for ITER candidate materials under the heat shocks. After upgrades the code was used for simulation of the amount of carbon dust particles and of the distribution of their sizes. The code has been tested against available experimental data from the plasma gun MK-200UG and from the electron beam facility JUDITH.

CARBON REMOVAL IN STAINLESS STEEL WALL AND GRAPHITE SHEET BY USING OXYGEN GLOW DISCHARGE

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Carbon fiber composite (CFC) is used as a divertor material around at trace regions in ITER. CFC is significantly eroded, and then eroded carbons co-deposit with fuel hydrogens on the first wall. It is regarded that the co-deposited layer has an amorphous structure and contains a large amount of tritium, so that this layer has to be removed. Oxygen glow discharge is proposed as a method for removal of co-deposited layer. In order to apply this technique, the basic effect of oxygen glow discharge on erosion of carbon has to be investigated. For this purpose, in the present study, the effect of oxygen glow discharge on removal of carbon content in stainless steel surface with a carbon concentration of 40 at.% was investigated. Similar experiment was conducted for a graphite sheet.

Glow discharge was conducted between SS liner cathode with a surface area of 7700 cm2 and copper anode placed in center of liner. Similar glow discharge was carried out in the same apparatus with a SS liner where a graphite sheet (ETB-10 with area of 300 cm2) was attached at the inner wall. Discharge gas was He and O2 mixture, where O2 partial pressure was adjusted in the range from 0 to 7%. Total pressure was kept 8 Pa before the discharge using a mass flow controller. The pumping speed was 220 cm3/s in N2 conversion. The discharge voltage was in the range from 200 to 300 V. The discharge time was taken 3 hours. The ion fluence on the wall was estimated as 1 x 10^18 ions/cm2 in this discharge. Amounts of desorbed gas species were quantitatively measured by a quadrupole mass spectrometer, and then a removal rate for carbon content was estimated.

In the case of SS cathode, desorptions of CO, CO2 and H2O were observed. The desorption amount of CO was largest. These desorption significantly depended on O2 mixture ratio. The CO desorption rate with the discharge of 7% O2 was approximately 7 times larger than that of 0% O2 mixture. Similar tendency was observed for CO2 and H2O desoptions. The desoptions of CH3, C2H5 and CH3O were observed during the discharges. The desorption amounts of these gases, however, little depended on the O2 mixture ratio. In the case of the discharge with the graphite sheet, the desorption amounts of CO and CO2 were several times larger than those in the case of SS liner. In the discharge without O2 mixture, desorption rates of CO and CO2 were comparable to those in the discharge without the graphite sheet.

The ratio of supplied oxygen gas to desorbed CO and CO2 was estimated. Approximately 30% of the supplied oxygen was consumed as CO and CO2 desorptions in the discharge with SS liner and 7% O2 mixture. In the case of discharge with graphite sheet and 7% O2 mixture, approximately 60% of supplied oxygen was consumed as CO and CO2 desorptions. Carbon removal rate in the graphite sheet was estimated as 3 x 10^-6 g/cm2 in 3hr discharge.

In the present study, it was seen that CO and CO2 desorptions were major processes for the carbon removal in the oxygen glow discharge. In addition, it was found that more than a half of supplied oxygen was consumed for the carbon removal in the graphite sheet. The present experiments clearly showed that oxygen glow discharge was very useful to reduce the carbon at the wall surface. In a case of amorphous carbon film produced in ITER, the carbon removal ratio of supplied oxygen may be larger than this value.

PS1-2101

CO-DEPOSITED CARBON FILMS PRODUCED IN THE VICINITY OF LOCAL ISLAND DIVERTOR IN THE LARGE HELICAL DEVICE

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In the Large Helical Device (LHD), Local Island Divertor (LID) experiments have been conducted from the 8th experimental campaign (2004).
Hydrogen gas was employed in the LID discharges. The divertor head made by carbon fiber composite (CFC) receives relatively high particle and heat fluxes, and then eroded carbons deposit on the wall with the hydrogen. In ITER, one of major concerns is an evaluation for fuel hydrogen retention of co-deposited carbon dust or film, since a tritium inventory in the carbon film is presumed to be significantly large. However, the fuel hydrogen retention of carbon films produced in fusion devices has not been sufficiently investigated so far. In order to investigate the fuel hydrogen retention of co-deposited carbon dust, material probes were installed in the vicinity of the LID head in the 8th and 9th experimental campaigns. A number of the LID discharge was approximately 500 in each campaign. The samples were placed close to LID head (at the wall 20 cm far from a top of LID head), and far from LID head (at the wall far from 87 cm). After the campaign, the samples were extracted, and the surface morphology, depth profile of atomic composition in carbon deposited layer and hydrogen retention were investigated using scanning electron microscope, Auger electron spectroscopy and thermal desorption spectroscopy, respectively.
The surface of the probe at the position far from the head was very smooth, but the protuberant parts with a submicron size were observed for the probe at the position close to the head. This difference might have been caused by the deposition process, which depends on the angle of eroded carbon to the probe surface. The smooth surface might have been caused by the deposition of hydrocarbons and/or low energy carbon atoms. In every carbon film, the carbon concentration was close to 100 at.% and oxygen concentration was only 1 at.%. The thickness of carbon film on the probe placed close to the head was larger (500 - 700nm) than that placed far from the head (160 - 200nm).
Most of retained hydrogen desorbed in form of hydrogen molecular during the thermal desorption measurement. The fraction of hydrogen desorbed in form of methane was very small, only several percents. Thermal desorption spectra in the samples near the head have a peak around at 1050 K, which is similar with that in graphite. However, the desorption peaks in the samples far from the head were observed at low temperature regime, as high as 100 K lower compared with the samples near the head. This suggests that the carbon structure at probe positions far from the head is clearly different with that of graphite, i.e., binding state of hydrogen differs from that of graphite. In addition, the hydrogen concentration in the samples far from the head was approximately double of those in the samples close to the head.The present results show that the hydrogen concentration of co-deposited carbon film becomes high at the position far from the plasma and the hydrogen desorption behavior also depends on the relative position to plasma.
This work was supported by the Collaboration Research Program for Large Helical Device (NIFS04KOBS005) in National Institute for Fusion Science, and partly supported by the JSPS-CAS Core University Program in the field of Plasma and Nuclear Fusion.

THERMAL LOAD RESISTANCE OF EROSION-MONITORING BERYLLIUM MAKER TILE FOR JET ITER LIKE WALL PROJECT

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The ITER reference materials, beryllium (Be), carbon fibre composite (CFC) and tungsten (W), have been tested separately in tokamaks. An integrated test demonstrating both compatibility of metal plasma facing components with high-power operation and acceptable tritium retention has not yet been carried out. At JET, the size, magnetic field strength and high plasma current allow to conducting tests with the combination of the materials. Thus, the ITER-like Wall (ILW) project has been launched. In the project, Be will be the plasma-facing material on the main chamber wall of JET. To assess the erosion of the Be tiles, a Be marker tile was proposed and designed. The test samples which simulate the JET Be marker tile have been produced in MEdC, Romania in order to study the thermal load resistance of the JET Be marker (20 x 20 mm2 size with 30 mm height). The marker tile sample consists of bulk Be, high-Z interlayer (2-3 um Ni coating) and 8-9 um Be coating. Thermionic Vacuum Arc (TVA) techniques based on the electron-induced evaporation have been selected for this purpose. In the present work, the global characterization of the maker tile samples and thermal load tests were performed.

After the pre-characterization (microstructure observation by scanning electron microscope and elemental analysis by means of Wavelength Dispersive X-ray Spectroscopy and Energy Dispersive X-ray Spectroscopy), the thermal loading tests were performed in the electron beam facility JUDITH. The coating consisted of tiny platelets of ~0.1 um in diameter and localized larger platelets of 1 um in diameter. The surface and bulk temperature were observed during the tests. In the screening thermal load test, the samples were loaded to 6 MW/m2 for 10 s. The layers did not show any macroscopic damages at up to 4.5 MW/m2 for 10 s (45 MJ/m2). However, the coating delaminated and the maker was damaged when the thermal loading reached at 5 MW/m2 (~50 MJ/m2). Cyclic heat load tests were performed at 3.5 MW/m2 for 10s. The surface temperature increase of the marker tile sample was around 670 K over the whole 50 cycles. After the thermal loads, microstructure observation did not show significant modification, although, the localized larger platelets were eroded. The cross sections of the coatings before and after the loading will be discussed.

OBSERVATION AND MODELLING OF HYDROGEN ATOMIC AND MOLECULAR IONS ON DIVERTOR SIMULATOR

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In a divertor plasma of fusion reactors, vibrationally excited hydrogen molecules H2(v) persist in dissociation and ionization processes of the plasma volume. However, atomic and molecular processes with H2(v) molecules have not been reported clearly for high-density plasma. We have carried out the experimental observation and modelling of atomic and molecular ions in hydrogen high-density plasma on a divertor simulator,TPD-SheetIV[1]. The hydrogen plasma was generated at a discharge current of 30-100 A. Electron density and electron temperature were measured using a planar Langmuir probe, which were located 3 cm in front of the endplate. An omegatron mass analyzer, situated behind a small hole in the endplate with a differential pumping system, is used for analyzing ion species [2]. The relative densities of the molecular and atomic ions were determined from the collector current of the mass analyzer. To model the ion density in this experiment, a simple zero dimensional model is developed for solving the system of rate balance equations for ion and gas species. In the reactions involving H2, H2+, a ground-state vibrational temperature of hydrogen molecule is Tvib = 3000 K in the model. The ion density ratio of H+ is larger than that of H2+ or H3+ in the low gas pressure (< 6 mtorr) at the discharge current Id of 50 A in the hydrogen plasma. On the other hand, the ion density ratio of H3+ rapidly increases from 0.1 to 0.5 with increasing gas pressure P and saturated up to P = 10 mtorr. From a zero-dimensional model using the relevant rate balance equations, the calculated molecular ion densities of H2+, and H3+ was found to predict the observed dominant ion density ratio, demonstrating the importance of vibrational temperature of hydrogen molecule.

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DAMAGE TO TUNSTEN MACRO-BRUSH TARGETS UNDER MULTIPLE ELM-LIKE HEAT LOADS. EXPERIMENTS VS. NUMERICAL SIMULATIONS AND EXTRAPOLATION TO ITER

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Operation of ITER at high fusion gain is assumed to be the H-mode. A characteristic feature of this regime is the transient release of energy from the confined plasma onto PFCs by multiple ELMs (about 104 ELMs per ITER discharge), which can play a determining role in the erosion rate and lifetime of these components. The expected energy heat loads on the ITER divertor during Type I ELM are in range 0.5 – 4 MJ/m2 in timescales of 0.3-0.6 ms.
Tungsten macrobrush armour (W-bruhes) is foreseen as one of plasma facing components (PFC) for ITER divertor and dome. During the intense transient events in ITER the surface melting, melt motion, melt splashing and evaporation are seen as the main mechanisms of W-erosion. The expected erosion of the ITER plasma facing components under transient energy loads can be properly estimated by numerical simulations validated against target erosion of the experiments at the plasma gun facility QSPA-T. Within the collaboration established between EU fusion programme and the Russian Federation, W-brush targets (produced either from pure tungsten or tungsten with 1% of La2O3) manufactured according to the EU specifications for the ITER divertor targets, have been exposed to multiple ITER ELM-like loads in plasma gun facilities at TRINITI in the range 0.5 – 2.2 MJ/m2 with pulse duration of 0.5 ms. The measured material erosion data have been used to validate the codes MEMOS and PHEMOBRID.
Numerical simulations, including 3D-simulations (codes MEMOS and PHEMOBRID), carried out for the conditions of the QSPA-T experiments with heat loads in the range 0.5 – 2.2 MJ/m2 and the timescale 0.5 ms demonstrated a rather good agreement with the data obtained at the plasma gun facility QSPA: melting of brush edges at low heat loads, intense melt motion and bridge formation caused by the Rayleigh-Taylor instability at heat loads Q>1.3 MJ/m2. The melt splashing generated by the Kelvin-Helmholtz, and Rayleigh-Taylor instabilities are analyzed. To eliminate a large damage to brush edges, large melt splashing, and intense formation of the bridges between brushes optimization of the W-brush geometry is carried out.
Damage of W-brush targets with both standard and optimized geometry was simulated for ITER ELM-like heat loads in the range 0.5 – 3.5 MJ/m2 and the timescales of 0.3-0.6 ms using the codes MEMOS and PHEMOBRID. Transformations of the brushes under multiple heat loads were also analyzed in the paper for different load conditions.

EXPERIMENTAL STUDY OF MHD EFFECTS ON TURBULENT FLOW OF FLIBE SIMULANT FLUID IN A CIRCULAR PIPE

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Experimental studies of MHD turbulent pipe flow of Flibe simulant fluid have been conducted as a part of US-Japan JUPITER-II collaboration. Flibe is considered as a promising candidate for coolant and tritium breeder in some fusion reactor design concepts because of its low electrical conductivity compared to liquid metals. This reduces the MHD pressure drop to a negligible level; however, turbulence can be significantly suppressed by MHD effects in fusion reactor magnetic field conditions. Heat transfer in the Flibe coolant is characterized by its high Prandtl number. In order to achieve sufficient heat transfer and to prevent localized heat concentration in a high Prandtl number coolant, high turbulence is essential. Even though accurate prediction of the MHD effects on heat transfer for high Prandtl number fluids in the fusion environment is very important, reliable data is not available. In these experiments, an aqueous solution of potassium hydroxide is used as a simulant fluid for Flibe. This paper presents the experimental results obtained by flow field measurement using particle image velocimetry (PIV) technique. The PIV measurements provide 2-dimensional 2-velocity component information on the MHD flow field.
The test section is a circular pipe with 89 mm inner diameter and 7.0 m in length, which is 79 times pipe diameter. This relatively large diameter pipe is selected in order to maximize the MHD effects measured by Hartmann number (Ha=BL(sigma/mu)1/2), and to allow better resolution of the flow in the near-wall region. The test section is placed under maximum 2 Tesla magnetic fields for 1.4m of the axial length. The hydrodynamic developing length under the magnetic field is expected to be 1.2 m. In order to apply PIV technique in the magnetic field condition, special optical devices and visualization sections were created. PIV measurements are performed for Re = 11600 with variable Hartmann numbers. The turbulence statistics of the MHD turbulent flow are calculated from 5000 samples of the vector maps obtained by PIV, and the results are compared with the available direct numerical simulation data. The instantaneous fluctuating velocity maps are also examined in order to improve understandings of the spatial structure of the MHD turbulence.

THE SANDIA PLASMA MATERIALS TEST FACILITY IN 2007

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The Plasma Materials Test Facility (PMTF) is now in its third decade of operation at Sandia National Laboratories. Recent upgrades to the facility have replaced the previous 30kW electron gun with EB60, which has a 60kW gun and associated grid control and power supply. In 2007 this system is being used in the testing of small mockups to assess the joining of beryllium and CuCrZr and the joining of CuCrZr to 316LN-IG to be used in the fabrication of US first wall panels for ITER. The EB1200, our 1.2 MW dual gun electron beam system, is being fitted with hardware to accept beryllium-armored first wall quality mockups to be tested in the summer of 2007. These water-cooled mockups will use PMTF water cooling system which also has the capability of operation at high temperature and high pressure. In addition, helium cooling and liquid metal cooling are also possible for samples on EB60. PMTF also has developed a liquid lithium loop (LIMITS) that was used to test free surface lithium flow through a magnetic field that simulated the a divertor target in NSTX, and in 2007 this work is being extended to include a lithium wetting test stand for use with liquid lithium divertor components being developed for NSTX in collaboration with the University of California, San Diego and the Princeton Plasma Physics Laboratory. PMTF also has a water loop with cyclic heating in which we are studying the potential for the evaluation of joining flaws using the phase lag of infrared images. This paper will provide an updated description of the PMTF and some ongoing activities.

PS1-2110

DNS OF TURBULENT HEAT TRANSFER UNDER A UNIFORM MAGNETIC FIELD AT HIGH REYNOLDS NUMBER

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In recent developments in nuclear fusion research, certain design concepts for liquid breeder blankets for nuclear fusion reactors use molten salts, such as FLiBe, as coolant material. The mean velocity of liquid coolant material in a reactor is strongly influenced by magnetic field, and hence it is important to study the turbulent magnetohydrodynamic (MHD) flow behavior for an applied magnetic field perpendicular to the main flow. Furthermore, because the flow characteristics of coolant at high Reynolds number are assumed to be different from the usual turbulent MHD flow, it is important to investigate the flow under a magnetic field where the Reynolds number is high. A direct numerical simulation (DNS) of turbulent heat transfer with high Reynolds number has been carried out to show the effects of magnetic field. In this study, the Reynolds number for channel flow based on bulk velocity , viscosity , and channel width was set to be constant; Reb = 45818. A uniform magnetic field was applied in the direction of the wall normal. The values of Hartman number Ha were 32.5 and 65. A constant temperature was applied to the wall as a thermal boundary condition. Prandtl number of the working fluid was assumed to be 0.06. The number of computational grids used in this study was 1024 x 1024 x 768 in the x-, y- and z- directions, respectively. The turbulent quantities such as the mean flow, mean temperature, turbulent stress, and turbulent statistics were obtained by DNS. Moreover, the large-scale turbulent structure about temperature field will be presented at final paper.

CRITICAL HEAT FLUX EXPERIMENTS USING SCREW TUBE UNDER DEMO-RELEVANT COOLING CONDITION

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As part of development of Plasma-Facing Components (PFCs) for fusion machines, JAEA has been developing high performance cooling tubes with pressurized water flow. Along this line, a cooling tube with a helical triangular fin on its inner surface has been proposed recently for application to a DEMO reactor to enhance heat removal. Since the fin can be machined by a simple mechanical threading, this tube is called as a screw tube. In our previous experiments, it was reported that heat removal performance of the screw tube with pressurized water at room temperature is twice as high as that of a smooth tube. Divertor cooling conditions in DEMO design in JAEA are envisaged to be at the pressure of 4MPa and the outlet temperature of ~ 200oC to improve thermal efficiency of power generation.
In the present study, effect of subcooling on critical heat flux (CHF) of the screw tube under DEMO-divertor-relevant condition has been investigated. A test sample is the screw tube made of OFHC-Cu with M10 of 1.5-mm-pitch, which achieved the highest CHF. The M10 threads are directly shaped in the tube with the outer diameter of 12 mm. The minimum wall thickness of each tube is 1 mm. Inlet temperature and local pressure of cooling water are ranging from 35 to 180oC and at 4MPa. Mass flow rate ranges from 0.22 to 0.66 m/s reduced to axial flow velocity from 4 to 12 m/s at 35oC. In CHF testing, the test sample is heated by using hydrogen ion beam. Incident heat flux at the sample position has a Gaussian profile and its maximum value ranges from 8 to 60MW/m2. Temperature excursion of the tube wall after the start of burnout is detected by thermocouples inserted in the tube wall and an IR camera observation.
Incident CHF (ICHF) defined at the outer tube wall depends the strongly on the mass flow rate and the degree of subcooling, Tsub (= saturated water temperature - bulk water temperature, Tbulk) almost linearly within the experimental conditions. At the mass flow rate of 0.56 kg/s corresponding to 10m/s at room temperature, ICHF decreases from 48 MW/m2 at Tsub = 200K (Tbulk = 50oC) to 26MW/m2 26 MW/m2 at Tsub = 60K (Tbulk = 197oC).
Although temperature rise of the cooling water with 140K leads to reduction of ICHF by almost half compared with those values at room temperature, the ICHF values of the screw tube remained more than double values of the smooth tube at the same cooling conditions.
This result encourages us to examine further applicability of the screw tube to DEMO divertor cooling structure.

PS1-2112

HEAT TRANSFER AUGMENTATION OF A CIRCULAR PIPE FLOW USING NANO-PARTICLE LAYERS

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For the advanced fusion reactor FFHR2 (Force Free Helical Reactor) that has been proposed by NIFS, molten salt Flibe ($LiF:BeF_2=64:36$) breeder blanket system is selected because of Flibe's features such as chemical stability, low-pressure operation and low electric conductivity. The Flibe is however high Prandtl number fluid since it has high viscosity and low thermal conductivity. Therefore its heat transfer performance is low compared with liquid Li or Pb-Li.
In addition to heat removal of $1MW/m^2$ on the first wall, electrolysis of molten salt due to MHD effect will take place under high flow rate condition. This indicates that heat transfer enhancement under low flow rate is essential for the Flibe blanket system.
In our laboratory, heat transfer characteristics of molten salt HTS ($KNO_3:NaNO_2:NaNO_3$ =53:40:7), have been evaluated, which is used as a simulant fluid of Flibe from the points of view of Be's toxicity and similar Prandtl number. In this paper, we adopt nano-particle layer method to form nano~micro scale structure on a heating surface using an acid or an alkali includes nano particles. There exist two methods to form nano particle layer. One is NPLS (Nano Particle Layer Structure) method which uses a chemical etching with an acid or an alkali including copper-oxide nano-particles. The other is FP (Fine Particle) method which employs electroless plating with inorganic metal salt solution.
At first, immersion experiments of NPLS or FP layers into melted HTS shows that erosion of the FP sample is much less than that of the NPLS sample. Furthermore, a forced-convention heat transfer experiments with a circular tube whose inner surface has the nano-particle layer by the FP method is carried out in a large molten salt circulating loop named as TNT loop. Results show that average nusselt numbers of the circular tube flow are about 1.3 times higher than that of a bared tube in the range of $3000<Re<13000$ and $13<Pr<27$.At the same time, immersion experiment of the FP layer under HTS flow is carried out to find that the FP's surface is oxidized by HTS and turned into crystal structure.
Secondary, in order to figure out optimal structure of the nano-particle layer, the heat transfer and pressure drop characteristics are evaluated for four tubes each of which has different surface structure. Those tubes were made by changing electroless plating time as a parameter (7, 10, 13,16min). The averaged nusselt numbers of the tube whose plating time is 16min become the highest while the pressure drops of the tube whose plating time is 7min are the lowest. Finally, the tube whose plating time is 13 min indicates the best performance from the view point of nusselt number ratios of the FP's tube to the bared circular tube for equal pumping power.

HEAT TRANSFER ENHANCEMENT IN SPHERE-PACKED PIPES UNDER HIGH REYNOLDS NUMBER CONDITIONS

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In recent design of LHD-type helical reactor FFHR, the first wall is expected to be exposed to the high heat load of almost 1.0MW/m2, which is removed by high temperature molten salt Flibe flow. The Flibe is a mixture of LiF and BeF2 and has advantages in high heat capacity, reduction of MHD-pressure drop due to its low electric conductivity. The Flibe blanket system, however, needs heat transfer enhancement under high heat flux since the Flibe is categorized as a high Prandtl number fluid. A Sphere-Packed Pipe (SPP) has been proposed as one of heat transfer enhancement techniques for the high Prandtl number fluid.
The matrix of SPP is composed of a number of spheres. The fluid is mixed in the process of passing through the complicated flow channels, which leads to high heat transfer performance. In addition, heat conduction between each sphere and a heating wall contributes to the enhancement of heat transport to the center of pipe, which is called fin-effect. However, the complicated structure causes relatively large pressure drop, which means it necessary to exactly solve the trade-off between the heat transfer enhancement and pumping-power increase in order to optimize the design.
Although several papers have been published relating to forced-convection heat transfer in SPPs, most of the studies have been performed under low Reynolds number regimes. In this study, therefore, the pressure drop and the heat transfer characteristics of the SPP flow are evaluated under high Reynolds number for different diameter ratios of the pipe to the sphere.
A test section is made of a stainless pipe with the diameter of 56 mm and acrylic spheres. The diameters of packed spheres are 18.5mm, 25.0mm, 27.6mm and 42.7mm, respectively. Water is employed as a working fluid. The pipe wall of 600mm length is uniformly heated by Joule heating.
Experimental results show that the pressure drop in the SPP flows approximately corresponds to the values between Ergun's correlation and drag model. The empirical correlation for averaged Nusselt number is proposed. In the full paper, the heat transfer performance in the SPP flows taking a pumping power into consideration will be compared with the other passive heat-transfer-enhancement techniques to discuss the possibility of the SPP system for the first wall cooling. Furthermore, the temperature distribution of the heating wall is visualized by using an infrared-ray thermography, which clarifies existence of both the high and low heat transfer areas depending on the packing structure.

PS1-2118

SIMULATION OF DISRUPTIONS ON NEUTRON IRRADIATED BERYLLIUM

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Thermo-mechanical and thermo-physical degradation of plasma facing materials caused by neutron irradiation, are important issues for the operation of next step fusion devices (ITER, DEMO, ...). Beside high thermal fatigue loads, due to the pulsed operation of tokamaks, the plasma facing materials suffer several types of transient loads which may lead to a heavy damage on these materials. These effects may even be deteriorated by the high energetic neutrons generated in the fusion process.
In the present study, the effect of disruptions on beryllium (which will be used as a first wall material for ITER) has been studied. Disruptions are simulated in the electron beam facility JUDITH by high energetic pulses of up to 100 MJ/m2. Under these loads, the beryllium surface may roughen combined with the forming of cracks. At higher power densities, a melt layer will form and finally the molten material is ejected or evaporated. After irradiation embrittlement an additional contribution of brittle destruction may occur.
In order to quantify the effect of disruptions depending on the grade of neutron irradiation and on the power density, the following diagnostic methods have been used:
* measurement of current through the sample to determine the exact amount of deposited energy
* weight loss measurement to quantify the amount of eroded material (weighting of each sample before and after the heat load experiment).
* surface micrography (by means of an image scanner located in the hot cell),
* laser profilometry to characterize the depth and the shape of the generated craters,
* hot metallography.
During the experiments, a special problem arises from the fact that during the neutron irradiation beryllium transmutes to tritium. This tritium is bound in the beryllium matrix, but during the heating of the samples, the tritium may be set free and through the vacuum pump it may be released to the environment. In order to avoid and to quantify this release of tritium, a special tritium trap has been constructed.
In this tritium trap, the gas is pumped by means of a metal bellows pump through a catalyst tube filled with copper oxide. At a temperature of 300°C, the tritium is oxidized to HTO. This HTO is lead through gas washing bottles filled with water. Here approximately 98% of the released tritium is caught. The temperatures in the process are controlled by thermo couples, and the tritium content is controlled by a tritium gas monitor.

AN EXPERIMENT ON THE CONCEPT OF ACTIVE RECYCLING CONTROL USING MOVING SURFACE PLASMA FACING COMPONENTS

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Steady state operation of reactor grade plasma requires capabilities of handling a tremendous amount of exhaust particles. It is extremely difficult to provide sufficient particle removal capabilities, employing commercially available pumps. Moreover, the application of wall conditioning techniques will also be limited in steady state fusion devices due to surface saturation phenomena. To resolve the steady state particle control issue the concept of moving-surface plasma-facing component (MS-PFC) was proposed about a decade ago and the proof of principle experiments were successfully performed using a rotating-drum MS-PFC test unit set in the laboratory plasma device.

In the present work, to evaluate the particle control capability of MS-PFC and its effect on the core plasma performance, a rotating-drum limiter has been put together and mounted on a Compact Plasma-surface interactions experimental Device (CPD). Similar to the laboratory test unit, the rotating drum is made of copper and is water-cooled. Also, two Li evaporators are positioned behind the rotating drum for continuous gettering. However, the critical difference is that the CPD limiter surface is coated with plasma-sprayed tungsten (0.5 mm thick) to prevent the damage from edge plasma bombardment. The exposure of the rotating target to the lithium vapor is controlled by a sliding shutter. The whole unit can be moved radially inside and outside the plasma with the help of a motor and bellow arrangement. The first-of-a-kind experiments have recently been done with this new limiter system exposed to RF plasmas in CPD. Typical discharge conditions are: RF =80kW; n_edge ~ 0.8x10^18 m^-3; T = 0.5 sec. Spectroscopic measurements have been done looking tangentially at the rotating drum surface from both top and radial locations. A CCD camera is used to monitor the interactions of plasma with the rotating surface. From the first experimental data of H_alpha light intensity, it is observed that hydrogen recycling reduces approximately by 7-10% infront of the rotating target, relative to no getter condition. The technical details of the MS-PFC unit along with the first experimental result will be presented in this paper.

EXPERIMENTAL LOOP FOR TESTS OF HELIUM-COOLED HHF COMPONENTS AT 600C/10MPA INPUT

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The Efremov Institute has completed the 1st stage of helium loop construction. This work has been carried out in collaboration with FZK.
The helium loop is intended for tests of divertor design options, which meet the requirements for the DEMO helium-cooled fusion reactor. These tests include selection of divertor materials and joining technology, optimization of the cooling regimes, thermal cycling of mockups to estimate the divertor component lifetime. It is also possible to use the loop in other areas of high-temperature helium nuclear technology (tests of first wall mock-ups, fission helium reactors relevant experiments). On the basis of the results of the systematic mock-ups tests it is planned to create the database for validation of the gas-dynamic and thermal mechanic codes used for simulation of processes in high-heat-flux components.
The helium loop is capable of testing the mock-ups at a nominal helium input temperature of 600oC and a pressure of 10 MPa. Maximum possible pressure losses in mock-ups amount to 0.5 MPa.
At this stage of works a stationary helium flow rate of 24 g/s is provided by oil-free membrane compressor. One more possible loop regime is by periodic gas pulses at 50 g/s and duration up to 120 s. The diagnostic system provides measurement of more than 40 gas and mock-up parameters such as pressure, flow rate and temperature, as well the surface temperature distribution by an infra-red camera. Mock-up heat loading is provided by the e-beam of the TSEFEY facility with an applied power of 60 kW. The testing results of single-finger mock-ups of the vertical target for the DEMO helium-cooled divertor are presented.
At the next stage the helium loop will be capable of attaining a flow rate of 150 g/s (using a helium circulating pump), which is sufficient for tests of nine-finger modules of the DEMO helium-cooled divertor. Some preliminary design simulations have been made in order to assess the hot pump performance for different pump types. As result of these simulations a vortex-type pump has been chosen and its design has been adapted to the helium loop requirements. Preliminary experiments have been carried out with a simplified model and drive in order to verify the accepted solutions.
The design solutions and R&D results of the activities at this stage of loop development are also presented.

DEVELOPMENT OF MODULAR HELIUM-COOLED DIVERTOR FOR DEMO BASED ON THE MULTI-JET IMPINGEMENT (HEMJ) CONCEPT: EXPERIMENTAL VALIDATION OF THERMAL PERFORMANCE

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A modular helium-cooled divertor design for the "post-ITER" demonstration reactor (DEMO) based on the multi-jet impingement concept (HEMJ) has been developed at the Forschungszentrum Karlsruhe [1]. The design goal is to accommodate a surface heat flux of at least 10 MW/m2 at an acceptable pumping power. This paper describes the thermal-hydraulic analyses and validation experiments performed in support of the HEMJ divertor design.
Both thermal-hydraulic and thermo-mechanical simulations were performed to support the original design optimization process [1]. The thermal-hydraulic analyses were performed using the FLUENT CFD software package; they showed that the HEMJ design can remove a heat load of up to 12 MW/m2 at an acceptable pumping power. Extremely high heat transfer coefficients were predicted (~30 kW/m2-K).
This experimental investigation has been undertaken to validate the results of the numerical simulations. A one-to-one scale test module that closely matches the reference geometry of the HEMJ design has been constructed and tested. Initial experiments have been performed using air as the coolant at different Reynolds numbers spanning the value for the actual helium-cooled HEMJ design. The experiments have been performed at heat fluxes of up to 1.0 MW/m2.
The temperature distributions and local heat transfer coefficients have been measured over a wide range of operational conditions. The experimental data have been compared with the results of a-priori analyses performed using the FLUENT CFD package with the same model options used in the original HEMJ divertor design calculations. Comparison between the model predictions and experimental data provides the means for assessing the suitability of the numerical model to the design of the HEMJ divertor, as well as other gas-cooled high heat flux components at fusion reactor operating conditions.
Future experiments will be performed using the same test module with helium cooling at prototypical Reynolds numbers and heat fluxes up to 2.0 MW/m2.

References:

1.P. Norajitra, et al., "He-cooled Divertor for DEMO: Experimental Verification of the Conceptual Modular Design," Proceedings of the ISFNT-7, May 22-27, 2005, Tokyo, Japan. Fusion Engineering and Design, 81(1-7), 2006, 341-346.

NUMERICAL INVESTIGATION OF A BRAZED JOINT BETWEEN W-1%LA2O3 AND EUROFER COMPONENTS

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The paper presents a numerical investigation of the plasticizing behavior of a conical brazed joint between W-1%La2O3 and Eurofer components of a He-cooled divertor finger under thermocyclic loading in the future DEMO fusion power plant.
A modular helium-cooled divertor design HEMJ (helium-cooled modular divertor concept with multiple-jet cooling) for the "post-ITER" demonstration fusion reactor (DEMO) has been developed at the Forschungszentrum Karlsruhe [1]. The design goal is to accommodate a surface heat flux of at least 10 MW/m2 at an acceptable pumping power.
A conical design of a brazed joint between tungsten and Eurofer structural parts of the HEMJ finger module has been investigated. Due to the large mismatch in the thermal expansion coefficients of tungsten (? ?4.6 *10-6 at 20°C) and Eurofer steel (? 10.4 *10-6 at 20°C), high thermal stresses occur in the joint when exposed to thermal cyclic loading between 20°C and 600°C, which could lead to the plasticizing of the materials in the joint region. A new design that withstands at least 1000 load cycles is based on the use of a conical joint between the tungsten and steel parts brazed to each other. To demonstrate the feasibility of the design, some steps have to be performed, such as numerical simulation, the choice of the braze material, study of the brazing technology, and the thermal cyclic tests of the finger mock-up.
In this paper the method of the numerical simulation as the first step will be described. For the stress calculations, the commercial software ANSYS was used, taking into account the material models, the thermal cyclic as well as the internal pressure loadings. The calculation results, in particular the plastic behavior of the brazed joint, will be discussed.

References:

1. P. Norajitra et al., "He-cooled Divertor for DEMO: Experimental Verification of the Conceptual Modular Design," Proceedings of the ISFNT-7, May 22-27, 2005, Tokyo, Japan. Fusion Engineering and Design, 81(1-7), 2006, 341-346.

ASSESSMENT OF THE HE-COOLED TEST DIVERTOR MODULE FOR ITER

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Within the EU power plant conceptual study (PPCS), a modular He-cooled divertor concept [1] is being investigated at the Forschungszentrum Karlsruhe to achieve a heat flux of at least 10 MW/m2. The intermediate-term goal of divertor development is the completion of a test divertor module (TDM) which is envisaged to be tested in ITER from 2020 onward. As a preparatory step, the possibility of performing such an experiment with a helium-cooled TDM in ITER has been assessed. The investigation covers e.g. checking whether the space available is sufficient and the experiment is compatible with ITER operation and RH procedures, assessing the thermohydraulic and the piping layout for helium cooling. The inlet and outlet temperatures of the helium coolant were chosen to be 600°C and 650°C, respectively. The complete CAD data set of an original ITER divertor cassette was used as a basis, with the main geometry comprising the inner (IVT) and outer (OVT) vertical targets, dome, and cassette body. Based on these data, the divertor target and helium coolant pipe were dimensioned taking into account the heat power to be removed and the difference between the helium coolant temperature (Tmean = 625°C) and the ITER water coolant temperature (Tmean = 125°C). Since a helium cooling system requires much more space than a water cooling system, the test of the helium-cooled divertor cannot be performed for the whole cassette. Hence, the tests of a He-cooled divertor concept need a compromise between a water-cooled unit (IVT, cassette body and dome) and a helium-cooled OVT, resulting in a combination of both cooling systems. This paper will also present further details about the choice of the ITER port suitable for testing the TDM and the access location for the helium pipes through the cassette body, the design of the attachment of the OVT to the cassette body, the optimisation of the reference divertor finger module, and the investigation of the helium-cooled finger modules in the OVT.

[1] P. Norajitra, R. Giniyatulin, N. Holstein, T. Ihli, W. Krauss, R. Kruessmann, V. Kuznetsov, I. Mazul, I. Ovchinnikov, and B. Zeep, Status of He-cooled divertor development for DEMO, Fusion Engineering and Design 75–79 (2005) 307–311.

S2-0001 (Invited)

STATE-OF-THE-ART 3-D NEUTRONICS ANALYSIS METHODS FOR FUSION ENERGY SYSTEMS

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Recent advances in radiation transport simulation tools enable an increased fidelity and accuracy in modeling complex geometries in fusion systems. Future neutronics calculations for design and analysis will increasingly be based directly on 3-D CAD-based geometries, allowing enhanced model complexity, reduced human effort and improved quality assurance.

Improvements have been made in both stochastic and deterministic radiation transport methodologies. To adapt the MCNP stochastic transport software, the translator approach allows CAD geometries to be converted from their native formats into standard input files, while the direct geometry approach uses computer graphics algorithms to perform the radiation transport on the CAD geometry itself. The former takes advantage of the efficiency of the native MCNP software without modifications while the latter permits the modeling of more complex surfaces. The ATTILA radiation transport package uses a finite-element formulation of the discrete-ordinate methodology to provide a deterministic solution on a tetrahedral mesh derived automatically from a CAD-based geometry.

All of these tools are being applied to a dedicated benchmark problem consisting of a 40 degree sector of the ITER machine defined only in a CAD-based solid model. The specific benchmark problems exercise the ability to use a CAD-based geometry to solve a range of fusion neutronics problems including neutron wall loading, deep penetration and narrow duct streaming. The results of this exercise will be used to validate/qualify these tools for use on ITER.

At the same time, many of these tools are being used to support the design of ITER components and other related fusion systems. UW has provided high-fidelity nuclear analysis of ITER first wall and shield modules identifying local effects of geometric features. ASIPP has used the MCAM tool to update and extend the existing ITER basic model and used it for neutronics analysis of the proposed Chinese ITER-TBM. FZK has used the McCAD interface programme to generate models of the Electron Cyclotron Resonance Heating (ECRH) launcher for integration into the standard ITER MCNP model. UKAEA have carried out design analysis of the RF antenna systems using both Attila and MCNP to determine a number of nuclear responses. Other systems being studied include the ARIES Compact Stellarator, IFMIF, and EAST.

The widespread use of these tools in the design and analysis of ITER and other fusion energy systems will enable a more accurate assessment of the nuclear response of individual components, leading to a reduction in design margins, improved overall performance, and new level of quality assurance (QA) since these new tools ensure that engineering and analysis models are consistent.

NEUTRONICS AND NUCLEAR DATA ISSUES IN ITER AND THEIR VALIDATION

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During the ITER R&D activities, the design of ITER has been supported by an intense experimental program at 14MeV neutron generators, dedicated to the validation of nuclear properties of critical components, such as the shielding blanket, and of relevant materials such as steels, tungsten and beryllium. The capability of codes and nuclear data to predict nuclear loads (nuclear heating, damage, activation) and resulting dose rates has been tested using both component mock-ups at neutron generators and the available measurements in real devices like JET. As the construction phase of ITER is starting, a rich although not yet complete data base of experimental tests is available as a result of this effort, which represents the basis for qualified and validated nuclear data and tools to be used in the design and safety analyses as required by the licensing procedure.

More recently, the experiments focused on the preparation for the Test Blanket Modules (TBM) program of ITER that aims at demonstrating the structural integrity under fusion-relevant loads, and their integral performance. From the neutronics point of view,the TBM tests aim at demonstrating the tritium breeding performance of the various blanket concepts and validate the capability of the neutronics codes and data to predict the nuclear responses with sufficiently high accuracy. For this reason, the neutronics TBMs will be equipped with appropriate neutron and tritium detectors. This may be the only opportunity for testing breeding blankets in a real fusion environment before the construction of DEMO.The success of such tests will depend, on one hand, on the quality of both the experimental techniques and the computational tools, i.e. on the level of uncertainties involved in the experimental and numerical analyses. On the other hand, it requires a detailed definition of objectives and design of measurements, taking into account the tests limitations in ITER, in terms of blanket coverage, neutron flux energy spectrum and fluence. Therefore, integral experiments on TBM mock-ups irradiated with appropriate neutron spectra are being carried out to validate, as much as possible, the computational tools and data prior to testing in ITER and, second, to develop and optimise the planned measuring techniques (in particular for the tritium measurements).

The present paper summarizes the results of the experimental activities aimed at validating the neutronics and nuclear data issues for ITER including, mainly the preparatory experiments on mock-ups of TBMs. Moreover, it describes the dedicated efforts needed to specify the neutronics tests and objectives in TBMs in ITER, considering the various concept configurations, the neutron flux conditions, the available measurement techniques, leading to a detailed design of the neutronics specific module, including instrumentation.

S2-0003 (Invited)

DEVELOPMENT OF ADVANCED BLANKET PERFORMANCE UNDER IRRADIATION AND SYSTEM INTEGRATION THROUGH JUPITER-II PROJECT

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Japan-USA Collaboration Program, JUPITER-II, has been progressed to study "The irradiation performance and system integration of advanced blanket" through 6 years plan for 2001-2006. The scientific concept of this program is to study the elemental technology in macroscopic system integration for advanced blanket based on microscopic mechanistic understanding. The program has emphasis on four research subjects as follows;
(1) Flibe system: Flibe handling, reduction-oxidation control by Be and flibe tritium chemistry, Thermofluid flow simulation and experiment and numerical analysis
(2) Vanadium /Li system: MHD ceramics coating of vanadium alloys and compatibility with Li, Neutron irradiation experiment in Li capsule and radiation creep
(3) SiC/He system: Fabrication of advanced composites with high thermal conductivity, Thermomechanics of SiC system with solid breeding materials
Neutron irradiation experiment in He capsule at high temperatures
(4) Blanket system modeling: Design-based integration modeling of flibe system and V/Li system, Multi-scale materials system modeling including He effect, Important recent results will be reviewed in the presentation.
As for (1) Flibe system, REDOX control of Flibe with beryllium was successfully demonstrated. It was confirmed that Be has enough solubility to reduce HF in the salt. A flow facility using KOH-water solutions to simulate key thermofluid parameters of Flibe was prepared. The velocity distribution obtained was in good agreement with the direct numerical simulation database.
As for (2)Vanadium /Li system, compatibility tests showed that high crystalline erbia coating on V-4Cr-4Ti was stable in static Li at 700C up to 1000hr. To optimize the composition of vanadium alloy layers for the multiple layers coating, compatibility of V-xCr-yTi alloys in static Li is under investigation. Thermal creep and irradiation creep tests using pressurized creep tubes were carried out in molten Li.
As for (3) SiC/He system, variety of thin-interphase chemical vapor infiltrated composites were designed and fabricated for irradiation in HFIR and for tailored thermal conductivity. Several advanced characterization technique for ceramics composites, including trans-thickness tensile strength and interfacial shear strength, were developed.
As for (4) blanket system modeling, work of the ARIES team in the US and FFHR team in Japan continued to contribute to JUPITER-II design interests. Evaluation of kinetics of He-vacancy cluster formation on ferritic steel was made by molecular dynamics simulations.

S2-0004

USE OF MCCAD FOR THE CONVERSION OF ITER CAD DATA TO MCNP GEOMETRY

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The program McCad developed at FZK provides a CAD interface for the Monte Carlo code MCNP. The use of CAD data ensures the consistency of the geometry model used in Monte Carlo neutronics analyses and the underlying design. This requires however an interface between CAD systems and neutronics Monte Carlo codes such as MCNP. McCad is able to convert CAD data into MCNP input geometry description and provides GUI components for modeling, visualization, and data exchange. It performs sequences of tests on CAD data to check its validity and neutronics appropriateness. Further, methods for the repair of CAD models and the completion of the final MCNP model by void geometries are implemented.

This paper describes the use of McCad to convert a 40 degree ITER torus sector CAD model provided by ITER to a suitable MCNP model and presents results of MCNP calculations performed to validate the converted geometry model. A dedicated CAD model for neutronics has been generated at ITER drawing office. It contains all significant components though some details have been suppressed and geometric simplifications were done in order to be compatible with the requirements of MCNP. The CAD model was then analyzed, tested, and corrected as required for the conversion process and the use in MCNP transport calculations. A fully functional MCNP geometry model was then generated by McCad. The conversion process does not introduce any approximations so that the resulting MCNP geometry is fully equivalent to the original CAD geometry. However, there is a moderate increase of the complexity measured in terms of the number of cell and surfaces.

The converted model has been validated by means of stochastic volume calculations which allow to compare the volumes of all MCNP geometry cells with the volumes provided by the CAD system. MCNP transport calculations were performed for specified nuclear responses such as the neutron wall loading, neutron fluxes at specified locations and the nuclear heating in specified components. The calculated responses are in a good agreement with results available from ITER.

S2-0011

ANALYSES OF FUSION INTEGRAL BENCHMARK EXPERIMENTS AT JAEA/FNS WITH FENDL-2.1 AND OTHER RECENT NUCLEAR DATA LIBRARIES

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Many integral benchmark experiments with DT neutrons have been carried out for nuclear data verification for fusion nuclear design at JAEA FNS;
1. Simple benchmark experiments : Neutron spectra, reaction rates, gamma-ray spectra, gamma heating rates, etc. were measured inside lithium oxide, beryllium, graphite, silicon carbide, vanadium, iron, SS316, copper, tungsten, etc. of simple geometry.
2. Time-of-Flight experiments : Angular neutron spectra above 50 keV leaking from slabs were measured for lithium oxide, beryllium, graphite, nitrogen, oxygen, iron, copper, lead, etc.
3. Breeding blanket experiments : Tritium production rates were measured in details inside lithium breeding layers in mockup assemblies for Japanese ITER test blanket module.
For a few years several nuclear data libraries have been newly released; JENDL-3.3 (2002 May), FENDL-2.1 (2004 Dec.), JEFF-3.1 (2005 May) and ENDF/B-VII.0 (2006 Dec.). It is essential to verify these libraries through analyses of integral benchmark experiments. Particularly validation of FENDL-2.1 is very important for fusion nuclear design. Thus we carried out a series of analyses for the benchmark experiments at JAEA FNS with FENDL-2.1, JENDL-3.3, JEFF-3.1 and ENDF/B-VII.0.
The Monte Carlo code MCNP-4C was used for this analysis. The ACE files supplied from JAEA Nuclear Data Center and IAEA Nuclear Data Services were adopted for JENDL-3.3 and FENDL-2.1, respectively. ACE files for JEFF-3.1 and ENDF/B-VII were produced with the NJOY99.161 code. Calculated results were compared with measured ones. They were also compared each other. Differences among the results with FENDL-2.1, JENDL-3.3, JEFF-3.1 and ENDF/B-VII.0 were not so large except for some experiments (silicon carbide, iron, lead, tungsten, etc.). In this symposium some typical results will be presented.

S2-0006 (Invited)

OVERVIEW OF SOLID BREEDER TBM CONCEPTS AND PROGRAMME FOR TESTING IN ITER

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The use of the solid breeder (SB) material technology offers attractive solutions for blankets of a first generation of Fusion Power Plant according to the possibility of a relatively simple design and operation of this component in reactor with high performances in term of thermal efficiency, tritium recovery and reduced dimensions. Almost all the ITER international parties have performed studies on such a concept and have presented an own blanket design for testing in ITER. Also if several different configurations of this kind of concept have been proposed during the past years, a set of common features can be identified that characterise this first generation concept, like the extern cooling of the breeder zone materials, the use of an independent low pressure helium flow for tritium recovery and Ceramic Breeder materials, mainly ternary Li-compounds, in form of a pebble bed. Furthermore, Ferritic/Ferritic-Martensitic steels at reduced activation grade have been selected for the structures; this choice dictates in strong way the performances of this kind of concept in term of minimum/maximum temperatures. All these concepts necessitate addition of large quantities (up to 4 times the amount of ceramic breeder) of beryllium or beryllium alloys as neutron multiplier in order to achieve a sufficient tritium breeder ratio with a reduced blanket thickness. Starting from this common base, different variants of this concept have been proposed. According to the coolant selection, two major classes of SB blanket can be identified, namely water cooled (SBWC) and helium cooled (SBHC) concepts. He cooling concepts have the advantage of a better chemical compatibility with the other materials and, in particular, with beryllium; water cooling concepts promise better thermo-hydraulics performances in term of cooling capacity of the first wall and a well proved cooling technology. Also the different arrangements of the breeder materials or the inclusion in the design of particular features typical of a reactor design (from DEMO or Fusion Power Plant Studies) lead to some variants for the reference blanket configuration. All these variants present, more or less, some different needs in part of the R&D programme.

A recent study in the frame of the Test Blanket Working Group aimed to identify general issues connected to this kind of breeder blanket concept and to assess particular needs related to the development of its different variants. The present paper presents and discusses the results of this study, outlining its outcomes for the implementation of an international co-operation programme and giving guidelines for the identification of a limited number of basic configurations suitable for being tested in ITER.

S2-0007 (Invited)

OVERVIEW OF LIQUID METAL TBM CONCEPTS AND PROGRAMS

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In support of the ITER Test Blanket Module (TBM) program, ITER party members have been focusing on the liquid metal blanket design concepts that have been extensively explored. With the use of reduced activation structural materials, we are designing to respective maximum allowable temperatures. For fusion power reactor designs, we will have to remove the first wall heat flux, breed adequate tritium for the D-T fuel cycle and achieve high coolant outlet temperature for high power conversion efficiency. After a period of assessment, we have selected liquid metal blanket concepts that can achieve these design requirements for respective DEMO designs. The objectives of our work are to establish the main characteristics of different proposed liquid metal blanket concept systems to be tested in ITER. We have identified compulsory design requirements from respective domestic DEMO strategies, completed the conceptual design of respective liquid metal TBMs, and identified necessary R&D programs. All liquid metal TBMs have the potential of meeting similar DEMO goals and requirements. All liquid metal TBM designs are to satisfy ITER safety requirements. Many R&D elements are common to a few designs such as the ferritic steel (FS) or V-alloy fabrication, thermal fluid MHD, FS/PbLi, FS/Li and V-alloy/Li compatibility, irradiation effects, tritium extraction, etc. With a well-coordinated ITER TBM program, different parties can supplement each other via collaboration. This paper will present respective designs and programs from the seven ITER party members.*

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S2-0008

OVERVIEW OF DESIGN AND R&D OF SOLID BREEDER TBM IN CHINA

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Tesiing of Breeding Blanket Modules (TBM) is one of ITER's important objectives. China is performing design and technology development of ITER TBMs based on the development strategy of fusion DEMO in China.
Solid breeder with helium-cooled test blanket module concept for test in ITER should be the basic option in China. The progress and status of CH HC-SB (China Helium-cooled Solid Breeder) TBM since 2004 are introduced briefly. Concept designs of HC-SB TBM and ancillary systems, test strategy for their tests in ITER, key R&D issues are summarized in this paper. An international collaboration in R&D, development and testing of TBMs are proposed.

Keywords: ITER; Test Blanket Module; TBM; Solid Breeder Blanket

S2-0009

THE HCLL TBM: PRESENT REFERENCE DESIGN, SYSTEM INTEGRATION IN ITER AND R&D NEEDS

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The Helium Cooled Lithium Lead (HCLL) blanket is one of the two European breeding blanket concepts under development for a DEMO power plant and for which mock-ups testing are planned in ITER in one of the 3 equatorial ports dedicated to Test Blanket Modules (TBM).
This paper gives an overview of the most recent developments in terms of TBM design, related analyses, fabrication developments, safety features, detailed testing objectives and priorities and corresponding planned test campaign. Required instrumentation, feasibility of its integration in the HCLL-TBM design, and needs of R&D on specific diagnostics will be discussed. Moreover, this paper addresses the issues concerning the interfaces of the HCLL-TBM system with ITER and the corresponding proposals of its integration in the ITER machine and buildings.
The addressed integration issues in ITER concern several aspects, such as TBM location and attachments in the Port Plug, PbLi ancillary circuit and part of the Helium Cooling System (HCS) components in the Port Cell, access allowance in the Port Cell region, HCS main components in the Torus Cooling Water System vault, piping in the shaft and galleries, required systems for Tritium extraction from He and from the PbLi purge gas in the Tritium building, space needs and specific Remote Handling facilities required for maintenance operations (repairing, Remote Handling disconnecting/connecting components, transport, storage, hot cell operations, …), and diagnostics Data Acquisition System in the control room.
For each of these integration aspects, general considerations will be given with a particular emphasis on the unsolved issues. Finally, the envisaged qualification campaign and developments needs for the installation of the HCLL TBM system for day-one of ITER operation, including the main identified milestones, will be summarized.

CURRENT STATUS OF DESIGN AND ANALYSIS OF KOREA HELIUM COOLED SOLID BREEDER TEST BLANKET MODULE

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It is known that the test of the capability of tritium self-sufficiency and the extraction of high grade heat using tritium breeding module concepts is one of the major ITER missions. This requires the development of test blanket modules based on the corresponding DEMO blanket design despite of the differences in operating conditions between DEMO and ITER. Two blanket concepts such as HCML (Helium-Cooled Molten Lithium) blanket and HCSB (He-Cooled Solid Breeder) blanket are considered as Korean Test Blanket Module (TBM) for ITER with the aim for verifying the capability of design and manufacturability of KO DEMO blankets.

In this paper, current status of design and analysis for Korea HCSB TBM is addressed. The design description, performance analysis results and key technologies are mentioned here. The key features of the design include: a) graphite neutron reflector used to reduce the amount of beryllium; b) simplified manifold for the helium coolant flow passage. The lithium silicate as solid breeder material, helium as coolant and tritium purge gas, low activation ferrite/martensite steel as structural material, beryllium as neutron multiplier are adopted in the design. Major performance analyses have been executed including TBM composition optimization, 3-D simulations and calculations of neutronics, thermal-hydraulic and mechanical analyses, reliability and safety of the blanket. Based on the design and analysis results, the licensing procedure of HCSB TBM will be proceeded in collaboration with ITER team. Finally the critical issues and the related R&D of the HCSB blanket design for the application in a DEMO reactor is mentioned.

THE MODIFIED RF CONCEPT OF CHC EXPERIMENTAL MODULE FOR TESTING ON H-H ITER PHASE

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The development of ceramic helium-cooled experimental module (CHC EM) is a part of RF concept for the Federal Government program to master the fusion nuclear energy and collaboration in the framework of international Test Blanket Working Group (TBWG).
The design decisions of CHC EM itself and it's ancillary systems should exposed by the combined tests under ITER operating conditions on H-H-phase and, possibly, on further operating stages. These design decisions or the corrected ones should be used as prototypes for the creation of DEMO blanket elements.
The modified concept of CHC EM (design and technological features) that will be tested on H-H-phase is described in this paper. The analysis results (including safety issues) are briefly presented too.

TRITIUM CYCLE SYSTEM FOR RF TBM AND THEIR SIMULATION IN NUCLEAR REACTOR

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In a frame of Russian DEMO Program and our participation in ITER activity a helium cooled lithium ceramic breeder with beryllium as a neutron multiplayer is one a variant of tritium breeding zone (TBZ) for a blanket. It is proposed to test the manufactured Test Blanket Modules (TBM) developed on a base of the researches in ITER and follow to design the blanket for DEMO. The program of TBM development includes RandD of TBZ materials and tritium technologies. To test TBZ of TM installation functional in-reactor investigations tritium-breeding models (RITM-F) was developed, designed and fabricated. RITM-F contains five parts: reactor assemble, tritium gas system, cooling system, control and monitoring systems and processing of experimental data. Two models of TBZ were tested at IVV-2M nuclear reactor and tritium extraction from lithium ceramic breeder materials was investigated. The results of the in-pile experiments are applied to development of TM TBZ. During operation some modifications were made, maintenance processes were developed to provide a radiation safety under operation. Results of in-pile experiments and RITM-F operations are presented in this paper. Keywords: Blanket, Breeding zone, Model, Tritium, In-pile testing.

PS2-1105

EXPERIMENTAL ESTIMATE OF TRITIUM PRODUCTION PARAMETERS FOR RF TEST BLANKET MODUL

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Tritium Breeding Ratio (TBR) is a most value among controlled fusion reactor parameters. One in a targets of Test Blanket Module (TBM) program is experimental investigation of the value. On the whole TBR can be submitted for consideration TBR=BTB/BTP (BTB - breaded tritium in blanket, BTP - burned tritium in plasma). To investigate a numerator of the formula a tritium production in breeding zone (BZ) of the TBM have to be measured under ITER plasma experiments. A tritium and neutron monitoring system with some lithium and neutron sensors is proposed. Lithium ortho-silicate and lithium carbonate and the neutron detectors fit the task. Differences isotope lithum-6 and lithium-7 can be applied. For delivery/withdrawal of the detectors into/from the BZ a pneumatic concept is suggested with using channels allocated in submodule. Channels pass through the submodule back wall and reach the attended area. These channels allow the insertion during the dwell time or operational pauses of activation foil and capsules with material probes. Capsules for the detectors and a channel for feed of the capsules in TBM before pulse and extraction after pulse are presented in this paper. Keywords: Test blanket module (TBM), Tritium, Lithium, Monitoring channel.

PS2-1106

BLANKET MANUFACTURING TECHNOLOGY : THERMOMECHANICAL TESTS ON HCLL BLANKET MOCKUP

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In the Helium Cooled Lithium Lead (HCLL) Blanket concept, the lithium lead plays the double role of breeder and multiplier material, and the helium is used as coolant. The HCCL Blanket Modules are made of steel boxes reinforced by stiffening plates. These stiffening plates form cells in which the breeder is slowly flowing. The power deposited in the breeder material is recovered by the breeder cooling units constituted by parallel cooling plates. All the structures such as first wall, stiffening and cooling plates are cooled by helium. Due to the complex geometry of these parts and the high level of pressure and temperature loading, thermo-mechanical phenomena expected in the “HCLL blanket concept” have motivated the present study.
The aim of this study, carried out in the frame of EFDA Workprogram, is to validate the manufacturing technologies of HCLL blanket module by testing small scale mock-up under representative operating conditions. The design and the manufacturing process of a cooling plate mock up are presented in a separate paper in this conference.
The experimental program is carried out on the DIADEMO facility at CEA Cadarache. This experimental device allows the coupling of a PbLi test section and a He cooling loop (pressure of 80 bar, maximum temperature of 500°C, mass flow rate of 150 g/s) taking advantage of synergies with the gas-cooled fission reactor R&D program. The first step of the test program aims at exposing a simplified HCLL cooling plate mock up in a fusion blanket environment (i.e in PbLi and cooled by pressurized Helium) to ITER relevant conditions with thermal transients simulating plasma shut down. Three thousands thermal transient cycles have been performed in about 3 months, without any external visible damages on the mock-up. Post tests analyses are underway in order to investigate the behaviour of the cooling plate mock-up.

PRELIMINARY LAYOUT OF THE HCLL TBM PIPING IN THE ITER PORT CELL

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Within the framework of the development of Test Blanket Modules (TBMs) to be tested in ITER, several activities on the integration of the module in the facility are under way. These activities address various integration issues, which include the space constraints and organisation, the positioning of the various systems and their impact on safety issues, and the maintenance and handling aspects.
The work presented here concerns the European Helium-Cooled Lithium-Lead (HCLL) TBM and consists in the definition of the main ancillary components in the Port Cell and the piping routes of the TBM connecting pipes, through the bioshield and the Port Cell up to the Vertical Shaft, in order to identify the main constraints dictated by the limited Port Cell volume.
The Helium and LiPb systems are concerned. The main components of the LiPb system, including the Tritium extractor, are located in the port cell and it is connected to the Tritium building where the ITER Tritium recovery system is located. The main components of the He system are located in the Tokamak Cooling Water System (TCWS) vault, reached through the vertical shaft, and therefore the hot He-pipes (up to 500°C) have to cross the port cell.
In particular, the pipes routes definition includes the pipes organisation, geometries and positions, the type and locations of the connections between pipes sections, required pipes thermal insulations and heating system (for LiPb), and the ways to handle the thermal expansions. The TBM shielding system, provided by ITER, is represented in the piping routes design.
All equipments present in the Port Cell are taken into account, by using symbolic representations: i) the LiPb components volumes are represented to figure out the piping connections; ii) the constraints due to the space sharing with another system in the Port Cell are identified with space limitation; iii) the Piping Integration Cask is defined and discussed to assess the corresponding space limitations.
The access for personal within the Port Cell had also to be considered as an additional constraint.

ASSESSMENT OF EM LOADS ON THE EU HCPB TBM DURING PLASMA DISRUPTION AND NORMAL OPERATING SCENARIO INCLUDING THE FERROMAGNETIC EFFECT

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Breeding Blankets will be tested in ITER by hosting Test Blanket Modules (TBM) in specially designed equatorial ports. Differently from the ITER shielding blanket modules, the TBM structural material is a low activation martensitic steel (like F82H or Eurofer) relevant for the future DEMO reactor development. This is a ferromagnetic steel with high saturation magnetic flux density (?1,9 T at 300 K). Therefore two categories of electromagnetic forces are expected on the TBM during all the ITER operating scenarios; the Lorentz Forces (LF) (caused by the interaction of the magnetic field with the eddy currents potentially flowing in the structure during the electromagnetic transients) and the Maxwell Forces (MF) that apply to a magnetizated body. The MF can be caused by two reasons: a misalignment of the body magnetization with the external field (that could potentially occur only during a fast Electro Magnetic (EM) transient) originating a torque and by a magnetic field not uniform in space originating net resultant forces directed toward the increasing external field. Due to the very high toroidal field in the region of the TBM the martensitic steel is highly saturated thus the non linearity of its ferromagnetic properties have a very negligible effect and the Eurofer in the TBM can be considered like a rigid permanent magnet with a specific magnetization equal to the saturation value.

To evaluate the effect of these loads on the TBM structure and attachment system, detailed electromagnetic analysis have been carried out and reported in this paper. Using ANSYS code several 3D models have been developed for both static and transient analysis. According to the ITER loads specification document a plasma current disruption of type II (linear decay in 40ms) has been identified as the most demanding event for the TBM equatorial position and its effects have been investigated using the EM zooming procedure (that allows a detailed modeling of a particular region of the whole machine maintaining the excitation information related to the structures that are not present in the zooming). LF on TBM have been evaluated and eddy currents distributions in the surrounding components have been stored for each time step. Using these currents as external loads in static analysis, the evolution of Maxwell forces during the plasma disruption have been investigated. While the maximum of resultant forces is reached at the End Of Burning time, just before the disruption beginning, the maximum of torque is obtained at the end of the disruption when the misalignment between the TBM magnetization (supposed to be rigid and thus anchored to the field direction at the disruption beginning) and the external field is maximum. At the end the effect of the toroidal field ripple on the MF has been evaluated taking into account the real geometric shape and discreteness of toroidal field coils. In spite of the small field perturbation produced by the ripple, the strong space gradient of this perturbation produces a very significant effect (an increase of about 50%) in the net resultant Maxwell force on the TBM.

A HELIUM COOLED MOLTEN LITHIUM TEST BLANKET MODULE FOR THE ITER IN KOREA

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Through a consideration of the requirements for a DEMO-relevant blanket concept, Korea (KO) has proposed a He Cooled Molten Lithium (HCML) blanket with Ferritic Steel (FS) as a structural material as part of the International Thermonuclear Experimental Reactor (ITER) program. The preliminary design and the performance of the KO HCML Test Blanket Module (TBM) are introduced in this paper. It uses He as a coolant at an inlet temperature of 300¡ÆC and an outlet temperature up to 400¡ÆC and Li is used as a tritium breeder by considering its potential advantages. Two layers of graphite are inserted as a reflector in the breeder zone to increase the Tritium Breeding Ratio (TBR) and the shielding performances. A 3-D Monte Carlo analysis is performed with the MCCARD code for the neutronics evaluation of the KO HCML and the total TBM power is designed to be 0.739 MW at a normal heat flux from the plasma side. From the analysis results with CFX-10 for the thermal-hydraulics evaluation, the He cooling path is determined and it shows that the maximum temperature of the first wall does not exceed 550 ¡ÆC for the structural materials and the coolant velocities are 45 m/sec and 8.2 m/sec for the first wall and breeding zone, respectively. The obtained temperature data was used in the thermal-mechanical analysis with ANSYS-10. The maximum von Mises equivalent stress of the first wall was 123 MPa and the maximum deformation of it was 3.73 mm, which is lower than the maximum allowable stress. And also, for the several accident scenarios such as a Loss of Coolant Accident (LOCA), a safety analysis is being performed.

PS2-1110

MONTE CARLO BASED SENSITIVITY AND UNCERTAINTY ANALYSIS OF THE HCPB TEST BLANKET MODULE IN ITER

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One of the important objectives of the nuclear testing of Test Blanking Modules (TBM) in ITER will be to check the capability of the neutronic codes and data to predict nuclear responses such as the Tritium production within sufficient accuracy. This requires the capability for assessing the uncertainties of the nuclear responses in the real three-dimensional reactor configuration as used routinely in Monte Carlo calculations.
A method to calculate sensitivities of Monte Carlo point detector responses has been previously developed and implemented in a local version of the MCNP code, called MCSEN.
The Monte Carlo based calculation of uncertainties of nuclear responses in the TBM of ITER, however, requires the capability to calculate sensitivities for responses by the track length estimator. Suitable algorithms based on the differential operator method were developed to this end and additionally implemented in MCSEN. This enables the efficient calculation of sensitivities for neutron fluxes and nuclear responses such as reaction rates in a geometry cell of an arbitrary 3D geometry. Sensitivities can be calculated to reaction cross sections, the material density and secondaries' angular distributions. Verification tests have been performed through the application to the TBM mock-up neutronics experiment conducted at the Frascati Neutron Generator (FNG) by comparing sensitivities calculated by the track length estimator with sensitivities calculated by the point detector.
This paper presents the first test application of the Monte Carlo sensitivity and uncertainty analysis to the TBM in ITER. To this end the standard 3D MCNP model of ITER (20 degree torus sector) with integrated TBM of the Helium-cooled Pebble Bed (HCPB) type was employed. Sensitivity profiles and integrated sensitivities were calculated for the total Tritium production in the HCPB TBM by using the track length estimator approach of MCSEN. The associated uncertainties of the Tritium production due to nuclear data uncertainties were assessed by making use of available co-variance data from different sources. The major contribution to the sensitivities was shown to come from the materials contained in the TBM, in particular the Beryllium neutron multiplier (positive contribution) and the breeder ceramics constituents Lithium6 (negative contribution) and Oxygen-16 (negative contribution). With the calculated sensitivity profiles and the available co-variance data, the cross-section induced uncertainties of the Tritium production have been assessed at a level of 2.8 %.

PS2-1111

THERMO-HYDRAULICAL AND THERMO-MECHANICAL ANALYSIS OF THE HCLL-TBM BREEDING UNIT

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The Helium Cooled Lithium Lead (HCLL) blanket is one of the two breeding blanket concepts under development in the European Union and for which Test Blanket Modules (TBM) are planned in ITER.
The HCLL-TBM consists of a steel box reinforced by an internal grid with Breeder Units inserted in the modular array (cells) defined by the grid. In each BU, cooling plates are used to extract the heat from the breeder/multiplier (Pb-17Li).
This paper presents the latest design of the HCLL-TBM according to the new dimensions of the space available for the module in the ITER equatorial ports. The performances of the proposed design are evaluated by means of detailed FE models of one BU and the corresponding box structure. The new models allow a better estimate of the heat recuperation effect between the different cooling elements, so that the coolant bulk temperature can be calculated along each flow path. As we show in the paper, this is necessary to achieve a correct design of the He cooling circuit. Detailed temperature and stress field distributions in each part of the structure have been obtained for nominal and accidental load conditions.
The results of the FE calculations are analyzed with respect to thermal-hydraulics and thermo-mechanical stresses in order to verify the compliance with the established design criteria (maximum temperature in the structural material and SDC-IC rules). Also, the operating parameters are discussed in view of the DEMO-relevance of the proposed TBM design. Finally, possible improvements and modifications of the current design are proposed.

PS2-1112

ELECTRIC FLOW COUPLING IN HCLL BLANKET MODULES

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A modular Helium Cooled Lead Lithium (HCLL) blanket concept, in which helium is used to cool the first wall and the breeder zones through cooling plates immersed in the liquid metal, has been proposed to be tested in the first phase of ITER operation as part of the Test Blanket Module (TBM) program of the European Union. In this blanket concept issues related to magnetohydrodynamics (MHD) have to be addressed in order to evaluate the feasibility of this blanket design.

A numerical study of the fully developed MHD flow in geometries related to the HCLL blanket has been performed. The investigated geometry consists of four breeder units in which He-cooled plates are inserted that subdivide the computational domain in sub-channels with high aspect ratio. The geometric features are chosen according to the characteristics of the experimental test section that is investigated in the liquid metal loop in the MEKKA laboratory of the Forschungszentrum Karlsruhe.

The effects of the orientation of the magnetic field on velocity and current distribution have been analyzed. In the case of pure toroidal magnetic field, jets with high velocity develop along the stiffening plates and a small increase of the velocity is also observed near the cooling plates. The electric coupling between breeder units is weak with currents that flow preferentially in tangential direction within the common stiffening plates. Inside the breeder units, the narrow ducts formed by the cooling plates are instead strongly coupled.

It has been found that the presence of a poloidal component of the magnetic field, required to confine the plasma, yields a stronger electric coupling between the breeder units and more complex current paths may be identified. Internal layers that develop along magnetic field lines appear and modify the velocity distribution in the breeder units compared to the case with pure toroidal magnetic field.

PS2-1114

HEATUP EVENT ANALYSES OF THE WATER COOLED SOLID BREEDER TEST BLANKET MODULE

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Water Cooled Solid Breeder (WCSB) Test Blanket Module (TBM) is being designed by JAEA as a primary candidate TBM of Japan. From the viewpoint of the safety, the TBM should be designed so that it does not damage the soundness of the vacuum vessel, the primary barrier for radioisotopes of the ITER. One of the major concerns on the safety of the TBM is temperature elevation due to coolant leakage into the neutron multiplier layer, beryllium, of the TBM. Since the chemical reaction of beryllium and water is an exothermic reaction and the reaction rate exponentially increases with the temperature increase, there is a possibility that the temperature of the TBM exceeds the maximum allowable temperature of its structural material. This paper describes the safety evaluation on the heatup events of the WCSB TBM and proposes the basic strategy to ensure safety, especially incorporating the chemical reaction between beryllium and water.

Failure Mode Effect Analysis (FMEA) has been carried out to select the severest heatup events of the WCSB TBM, followed by one-dimensional analyses to evaluate the selected events. The analysis model includes thermal conduction in the TBM, thermal radiation from the TBM to a common frame, and thermal radiation from the TBM first wall to the first wall of the opposite blankets (shield blanket etc.). The sequences of the selected events are shown as follows;

Loss of cooling of the TBM during plasma operation is assumed as an initial event. Temperature of the TBM totally increases, then a plasma disruption takes place when the temperature of the first wall armor reaches at a certain value, for example, its melting point of 1273 C. After the plasma disruption, temperature of the TBM decreases according to time and the event converges. However, if the pipe of cooling system in the TBM ruptures due to high temperature, chemical reaction between beryllium and water is activated and the TBM structure is possibly destroyed in the worst case. Therefore, the TBM should be designed so that no cooling pipe rupture can be guaranteed under the predicted highest temperature condition.

With respect to the ingress of coolant into the TBM, rupture of a pipe of cooling system inside the TBM during plasma operation is assumed. Coolant enters and fills in the TBM. Not only nuclear heat under normal operation, but also additional heat load due to the chemical reaction between beryllium and water should be taken into account. Transition of temperature of the TBM was calculated and the result showed that the temperature distribution is almost the same as that without the ingress of coolant. During normal plasma operation, temperature of beryllium layer is 600 C at most and the effect of the heat load due to the chemical reaction is much smaller than that of the nuclear heat. Even in the case of the ingress of the coolant during plasma operation, temperature of the WCSB TBM does not increase as long as the cooling system operates.

According to the results of these analyses, the following strategy of safety design is effective.

(a) WCSB TBM should be designed so that no cooling pipe rupture can be guaranteed after the loss of cooling and the heatup of the TBM.

(b) Cooling system of the TBM should be designed that it is not dependently terminated by the ingress of the coolant into the TBM.

HELIUM COOLED LITHIUM LEAD: ACTIVATION ANALYSIS OF TEST BLANKET MODULE IN ITER

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The Helium Cooled Lithium Lead (HCLL) Test Blanket Module (TBM) is one of the European blanket concepts selected to be tested in ITER as an important step towards the development of DEMO blankets. The EFDA organization has planned a work to study the problems related to the irradiation of a TBM in ITER, in order to optimize the test and to have a full extrapolation from ITER to DEMO conditions.

A very important issue concerning safety is the activation of HCLL materials induced by neutrons during operation. The flow of LiPb transports activated products outside the Vacuum Vessel, furthermore possible leak or accident can give consequences that should be analyzed to be ready for proper countermeasures to be taken. The products of activation can be chemically corrosive and proper maintenance has to be planned. Some troublesome radioactive nuclides, i.e. 210Po, represents high health risk. Hence activation calculation are necessary in support of the safety analysis. The design shall minimize the amounts of radioactive and toxic materials and the hazards associated with their handling.

A complete nuclear analysis was recently performed using the MCNP-4C Monte Carlo code, supported with FENDL-2 nuclear data library. The 3-D generic and most updated neutronic model of ITER machine has been completed with the insertion of the last HCLL TBM model in one of the equatorial port. The TBM model was described with sufficient detail to give enough reliability to the results turned to the designers. The neutron fluxes were calculated with MCNP in LiPb breeder units and in the Eurofer cooling plates at various positions inside the module for a D-T neutron yield rate of 1.77 1020 n s-1 (Pfus=500 MW). The activation calculations were performed using FISPACT (EASY 2005.1 package): activity, nuclear heating and contact dose rate were calculated inside the TBM using as input the neutron fluxes calculated by MCNP. Two irradiation scenarios were considered: 1) scenario representative for the irradiation of the TBM as scheduled for the high duty D-T phase of ITER with a total of 9000 neutron pulses over three (calendar) years period; 2) scenario characterized by an extended irradiation time according to the ITER M-DRG1 irradiation scenario (total first wall neutron fluence of about 0.3 MWa/m2) to arrive at a conservative estimate of the activity and afterheat production in case the TBM would be irradiated longer than initially assumed in the first case.

The results of the activation analysis in terms of activity, nuclear heat and dose rate at different times since shut-down till 106 y after at various radial, poloidal and toroidal positions are presented. The effect of impurities on LiPb and Eurofer has been investigated and discussed as well as the impact of the different irradiation scenarios on the activation of TBM.

HELIUM-COOLED PEBBLE BED TEST BLANKET MODULE ALTERNATIVE DESIGN AND FABRICATION ROUTES

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According to first results of the recently started European DEMO study, a new blanket integration philosophy was developed applying so-called multi-module segments [1]. These consist of a number of blanket modules flexibly mounted onto a common vertical manifold structure that can be used for replacing all modules in one segment at one time through vertical remote-handling ports. This principle gives new freedom in the design choices applied to the blanket modules itself. Based on the alternative design options considered for DEMO also the ITER test blanket module was newly analyzed. As a result of these activities it was decided to keep the major principles of the reference design like stiffening grid, breeder unit concept and perpendicular arrangement of pebble beds related to the First Wall because of the very positive results of thermo-mechanical and neutronics studies. The present paper gives an overview on possible further design optimization and alternative fabrication routes.

One of the most significant improvements in terms of the hydraulic performance of the Helium cooled reactor can be reached with a new First Wall concept. That concept is based on an internal heat transfer enhancement technique and allows drastically reducing the flow velocity in the FW cooling channels. Small ribs perpendicular to the flow direction (transverse-rib roughness) are arranged on the inner surface of the First Wall cooling channels at the plasma side. In the breeder units cooling plates which are mostly parallel but bent into U-shape at the plasma-side are considered. In this design all flow channels are parallel and straight with the flow entering on one side of the parallel plate sections and exiting on the other side. The ceramic pebble beds are embedded between two pairs of such type of cooling plates.

Different modifications could possibly be combined, whereby the most relevant discussed in this paper are (i) rib-cooled First Wall channels, (ii) U-bent cooling plates for the breeder units, (iii) manifold built from large vertical stiffening plates and welded horizontal stiffening plates as well as manifold plates welded between the vertical stiffening plates, (iv) attachment made from bending plates in combination with shear keys. In regard on the fabrication of cooling plates and stiffening plates wire cut EDM and electron beam welding techniques are considered, which could be used in case diffusion welding of the stiffening plates wouldn't be available in time for the first mock-ups.

PS2-1117

THERMOHYDRAULIC INVESTIGATIONS OF HELIUM-COOLED-PEBBLE-BED TEST BLANKET MODULE

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The Helium coolant flow in Helium-Cooled-Pebble-Bed Test Blanket Module (HCPB TBM) is characterized by (i) very intensive non-uniform first wall channel heating at the plasma side and (ii) a complex geometry of flow domain consisting of large coolant collectors with numerous flow obstacles and long narrow channels which are meandering in cooling and stiffening plates. In this regard, at the Institute for Reactor Safety of Forschungszentrum Karlsruhe two thermohydraulic aspects of HCPB TBM are currently under detailed investigation: The HETRA experiment in the framework of development activities for the first wall cooling and the GRICAMAN experiments in regard on the mass flow distribution among different and within individual components of the HCPB TBM coolant system.
The HETRA experiment has been motivated by corresponding three-dimensional (3D) numerical analyses which revealed significant effects of the asymmetrical heat loads on the cooling of the first wall. It was found, that the heat transfer coefficient in the first wall is ~15% lower than predicted by one-dimensional heat transfer evaluations based on Dittus-Boelter-like correlations and satisfactory cooling of the first wall can be achieved only with hydraulically rough channels. Additionally, due to strong temperature gradients in the cross-section of the first wall, the procedure for heat transfer evaluations applied in codes for stress analyses had to be modified in order to obtain reliable predictions of thermal stresses. The verification of the methods developed will be done on the basis of the results of the HETRA experimental campaign. I single first wall channel is tested in a Helium cycle at 80 bars, while the surface heat load is represented by a set of electrical heaters. From detailed temperature measurements in the structure the heat transfer in the first wall cooling channel can be determined.
The flow domain in the GRICAMAN experiment is defined to be the upper toroidal-poloidal half of TBM bounded at the outlets of the first wall channels, at the outlets of by-pass pipes and at the inlets of breeding units, i.e. involving one half of manifold 2, cooling channels in six horizontal and eight vertical stiffening grid plates, cooling channels within one and one half of manifold 3. Significant simplifications of the experimental facility and numerical models are achieved (i) assuming that the flow is adiabatic, (ii) replacing helium at 80bar and 3700C with air pressurised at 3bar and ambient temperature and (iii) representing complicated stiffening grid- and cap channels by simple pipes with the equivalent flow resistances. Preliminary to construction of GRICAMAN facility the following activities have been performed: (i) 3D simulations of the fluid flow in GRICAMAN domain and (ii) GRICAMAN sub-experiments for detailed investigation of the fluid flow in the real channels of vertical/horizontal stiffening grid plates.
In this paper a brief description of the HETRA and GRICAMAN test facilities will be given along with the detailed discussion of the first experimental results. The extensive CFD studies performed in the framework of both experimental campaigns will be described and results will be analysed in detail, including validation based on the experimental results.

PS2-1118

MANUFACTURING ASPECTS IN THE DESIGN OF THE BREEDER UNIT FOR HELIUM COOLED PEBBLE BED BLANKETS

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The breeding blanket programme has been in the focus of European fusion research for more than a decade. Recently, it has been driven by the EU Power Plant Conceptual Study (PPCS), investigating the potential of fusion energy in a future economic environment. On the way to the first commercial nuclear fusion reactor (DEMO) new studies for reactor in-vessel components have been initiated.
One central focus is the design and manufacturing of the blankets that have to ensure the breeding process to maintain the fuel cycle and are also responsible for the extraction of the main part of the reactor heat for power generation. Two kinds are established: One is the Helium Cooled Pebble Bed (HCPB) and the other the Helium Cooled Liquid Lead (HCLL) blanket. Both designs employ three different cooling plate assemblies. The outer, cooled U-shaped shell, namely the First Wall (FW), with two caps builds the blanket box. The structural strength of the blanket box is realized by integrating Stiffening Grids (SG) that separate the equally spaced Breeder Unit (BU) and allow the box, in case of faulted conditions, to withstand an internal pressure of 8MPa. The cooled SG constitute the side walls of the BU and are also cooled. The BU consists of a dedicated Cooling Plate (CP) assembly.
In present studies about the fabrication of Cooling Plates two kinds of diffusion welding processes are focused on. One is based on a Hot Isostatic Gas Process (HIP). The second is a uni-axial Diffusion Welding Process (DWP). In both cases the bond between the two halves of the cooling plate structure is reached by controlled pressure and heat cycles. Approaching larger, realistic scaled components the uncertainty of ensuring uniform process parameters across the bonding zone increases the risk of defect sources and, therefore, makes it difficult to guarantee the required bonding penetration. This study will present an alternative manufacturing strategy. The premises for this strategy are the reduction of volumetric joint technique and the substitution of joint technique by mechanical processing. This new proposal of CP manufacturing employs a conventional Spark Erosion (SE) process. Thus the design comprises straight cooling channels. Due to limits of the SE process the CP has to be assembled from three segments. For the joints Electron Beam (EB) welding was used to minimize thermal deformation and seam rise. The inspection of the parts and according thermo hydraulic calculations verify, that in spite of this design change, the hydraulic performance can be maintained. Operations like bending the CP in a U-shape to realise the CP assembly of a BU as well as qualification tests simulating realistic operating conditions and attest quality assurance are described.

MANUFACTURING EXPERIMENT OF A COOLING PLATE FOR A BLANKET BREEDER UNIT

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The realization of a future fusion power plant causes as a central task the manufacturing process of the breeding blanket box (BB) and the inserted breeder units (BU) containing the breeder material and the Beryllium. The BB and BU will be realized by cooling plates (CP) with a meandering internal cooling channel system. Such a CP can only be realized by joining two half pieces with half milled-in cooling channels. The most promising joining process for a CP with a large amount of welding area is the diffusion weld (DW) process. The more challenging CP would be the first wall (FW). But the FW will be the largest component of the BB and therefore it will be produced with a lot of difficulties. It is more promising to start with the manufacturing of small components of a BB so called mock ups as usual at industry. In the flow of such a process optimization the mock ups will be scaled to needed dimensions.

The current paper reports a DW-experiment which manufactures a CP in typical BU dimensions. The experiment generates experimental results of thermal transient behavior during diffusion weld process also. Such results are necessary adjusting the DW process to larger scale applications like a FW. The weld quality will be determined destructively by tensile and Charpy impact testing and compared by reference specimens taken from laboratory diffusion weld specimens without cooling channels.

PS2-1122

EXPERIMENTAL INVESTIGATIONS OF LIQUID-METAL MHD FLOWS IN A MOCK-UP OF A HCLL BLANKET

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Liquid metal magnetohydrodynamic (MHD) flows in a scaled mock-up of a helium cooled lead lithium breeding blanket is investigated experimentally in the MEKKA laboratory of the Forschungszentrum Karlsruhe. The experimental test section is a copy of a conceptual design study for a Test Blanket Module (TBM) for ITER, scaled down by a factor of two to fit into the magnetic gap of the available strong dipole magnet. As a model fluid the eutectic alloy NaK is used in the experiment, which allows performing experiments at room temperature. Moreover, its high electric conductivity allows reaching conditions, where electromagnetic forces dominate over viscous and inertia forces, as expected in applications for fusion. The experiment aims at determining the liquid metal flow rates in ducts formed by the cooling plates and investigating the electromagnetic flow coupling among the breeder units. The pressure drop is determined in breeder units, poloidal manifolds, expansions and contractions at the entrance and exit of the TBM mock-up, where there is an exchange of flow between circular pipes and rectangular poloidal manifolds. The present paper gives an overview of the project, describes the current status and shows first results.

MANUFACTURING OF A HCLL COOLING PLATE MOCK UP

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The European DEMO blankets and associated Test Blanket Modules (TBM) are made of a set of components cooled by flowing helium at 80bar pressure. Hot Isostatic Pressing (HIP) is one of the very few processes that allow manufacturing such components exhibiting complex cooling channels.
In HIP technology, the parts used to manufacture components with embedded channels are usually machined plates, blocks and tubes. Achievable geometries are limited in shape because it is not always possible to figure the channels by bent tubes. This occurs for example when channels present sharp turns, when the cross section of the channels is rectangular or when the rib between channels is so small that very thin tubes would be required. In these cases, bending is unpractical.
The breeder unit cooling plates of the Helium Cooled Lithium Lead (HCLL) blanket have eight 4x4.5mm parallel channels that run following a double U scheme. Turns are sharp and the wall thickness is small (1mm), so the manufacturing process described above cannot be used. An alternative process has been developed which has many advantages. It consists in machining grooves in a base plate, then closing the top of the grooves using thin welded strips, and finally adding a plate by HIP. There is then no need for the use of tubes with associated bending and deformation issues. The final component contains welds, but it must be stressed out that these potentially brittle zones do not connect the channels to the external surface because they are covered by the HIPed plate. Furthermore, the welds are homogenised during the HIP operation and further heat treatments.
This paper describes the design of a simplified cooling plate mock up and its fabrication using this so-called weld+HIP process. The thermal fatigue testing of this mock up is presented somewhere else in this conference.

A STUDY OF THE POTENTIAL INFLUENCE OF FRAME COOLANT DISTRIBUTION ON THE RADIATION-INDUCED DAMAGE OF HCLL-TBM STRUCTURAL MATERIAL

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Within the European Fusion Technology Programme, the Helium-Cooled Lithium Lead (HCLL) breeding blanket concept is one of the two EU lines to be developed for a Long Term fusion reactor, in particular with the aim of manufacturing a Test Blanket Module (TBM) to be implemented in ITER.
The HCLL-TBM is foreseen to be located in an ITER equatorial port, being housed inside a steel supporting frame, actively cooled by pressurized water. That supporting frame has been designed to house two different TBMs, providing two cavities separated by a dividing plate 20 cm thick.
As the nuclear response of HCLL-TBM might vary accordingly to the supporting frame configuration and composition, at the Department of Nuclear Engineering of the University of Palermo, a parametric study has been launched to investigate such an influence.
Previous works dealt with the dependence of the nuclear response of HCLL-TBM on the configuration of a homogeneous frame, the present one has been focused on the investigation of the potential influence of coolant distribution within the frame on the radiation-induced damage of HCLL-TBM structural material. To this purpose, a detailed parametric study of the HCLL-TBM nuclear response has been performed by means of 3D-Monte Carlo neutronic analyses to asses both the rates of displacements per atom and helium production within the structural material. A semi-heterogeneous model of the supporting frame, assuming a realistic coolant distribution, and a 3D heterogeneous model of the HCLL-TBM, taking into account 9% Cr martensitic steel (Z 10 CDV Nb 9-1) as structural material, have been set-up. Both the two models have been inserted into the existing 3D ITER-FEAT one, simulating realistically the reactor lay-out up to the cryostat and providing for a proper D-T neutron source.
The analyses have been performed by means of the MCNP-4C code, running a large number of histories for each one of them in such a way that results obtained are affected by statistical uncertainties lower than 1%. The results obtained are reported and critically discussed.

DESIGN OF THE INTEGRATION INTERFACE BETWEEN THE EU HCPB TBM AND THE ITER TBM PORT PLUG INCLUDING HOT CELL OPERATIONS FOR CONNECTION

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In the frame of the activities of the EU Breeder Blanket Programme and of the Test Blanket Working Group, the Helium Cooled Pebble Bed Test Blanket Module- (HCPB-TBM) System is developed. The TBM test schedule foresees four different campaigns for simulation of DEMO relevant conditions, campaign requires a dedicate TBM. Therefore a concept for TBM integration into ITER is designed with attention to simplify the mounting/dismounting operations. This paper presents the status of this concept with regard to the operations in hot cell required to install a new TBM into an equatorial TBM Port Pug (PP). This includes the establishment of the connection for the attachment, supply- and diagnostic lines in the environment of the interface (IF 1) between the TBM rear part and the PP backside shield.
The connection of IF 1 has to be designed to cope with a temperature difference between TBM and PP (~200 °K) and the EM-loads during normal operation and disruption scenarios. The reference attachment concept based on shear keys and flexible cartridges is revised to cope with new conditions on the load and at the interface to the PP. According to the latest results of EM analysis, a radial component of the Maxwell forces (due to the ferromagnetic structural material) has been identified as an additional challenging load for the attachment. Furthermore, the replacing operations at IF 1 are influenced by the design of the PP; the recent ITER proposal based on a removable back side shield allows access to the IF 1 from the periphery after the frame of the PP surrounding the TBM is removed. As for the mechanical attachment, the tools and operations for connection of the TBM supply lines (Helium-, Purge- and measurement lines for different purpose depending on the test schedule) are strongly influenced by the restrictions to access IF 1, too. Dismantling of the frame would allow direct access to the interface by e.g. orbital welding tools. The concept for connection of the TBM diagnostic lines does not foresee an interface between the TBM and the PP back side shield because of the very restricted space conditions. Therefore the diagnostic lines will be routed inside of a pipe which is attached to the TBM rear part. This instrumentation pipe is designed to penetrate the whole radiation shield up to the interface between the PP back side shield rear part and the Ancillary Equipment Unit (AEU). At this interface the diagnostic lines exit the instrumentation pipe by a feed through where they are connected to a multi plug which provides the connection to the Data Acquisition System. The vacuum boundary between the back side shield and the instrumentation pipe will be provided by a bellow.
After a consistent concept for the integration of the HCPB TBM in ITER has been developed, further investigation will be needed to develop tools and procedures which are required to install the TBM into the PP during the maintenance and refurbishment operations in the hot cell.

TEST BLANKET MODULE MAINTENANCE OPERATIONS BETWEEN PORT PLUG AND ANCILLARY EQUIPMENT UNIT IN ITER

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In collaboration between the FZK and KFKI-RMKI, in the frame of the activities of the EU Breeder Blanket Programme a concept for Test Blanket Module (TBM) integration, maintenance schedules and all required special purpose equipments has been developed.
During the first 10 years of ITER operation 4 different plasma scenarios will be used. Hence it will be possible to investigate the characteristics (e.g. tritium breeding performance) of different TBM concepts which will be installed during operation for the different phases of ITER operation in the equatorial ports #2, #16 and #18. In every port will be two TBMs accomodated, in the port #16 will be the the European Helium Cooled Pebble Bed blanket. In the different phases of ITER operation different TBMs will be used. Therefore a complex maintenance process is necessary for exchange the TBMs.
Two TBMs are mounted into one common frame, into a Port Plug (PP), which offers a standardised interface to the Vacuum Vessel (VV). It is cantilevered with a flange to VV Port Extension. This attachment system is the same in every equatorial port, so the exchange process of this structure with the TBMs are also standard operation of ITER.
Several components of the Helium cooling system of the EU breeder modules, valves, pipes, gas mixers, thermal sleeves, pipes for tritium extraction, measurement system, etc. All of them is integrated into the Ancillary Equipment Unit (AEU) which during operation will connect the port plug to the sub systems. The bigger part of the AEU is accomodated in the Port Cell and the rest part of it is penetrate to the interspace inside the bioshield and reach the back plane of the installed PP.
The remote handling operations for connection / disconnection of an interface between the PP of the EU-TBMs and the AEU are investigated with the goal to reach a quick and simple TBM exchange procedure. The current design of the EU-TBMs foresees up to 18 supply lines for both TBMs. These lines have to be connected here.
A new concept was worked out for fitting to each other these 18 pipe-ends and the opposite ones together in the same time in only one step before welding.
A special mechanical system is developed, which can transfer the robot from its storing place in the AEU in the port cell to the interface on 6 m distance. This mechanism has a minimized space requirement, and ables to send in the robot and all the tools stored in the tool-magazine together.
Weld seams can be made by orbital welding tool. The coolant is helium, so for eliminating the leak of helium it is of high importance to find a safe way for weld seam audit. The installation and removal of thermal insulations around the pipes at the interface is an additional requirement which has to be met by a special tool.

SAFETY CONSIDERATION OF TRITIUM SYSTEMS FOR CHINESE HCSB AND DFLL TBM

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The main goal of an ITER TBM is to test the feasibility of tritium production and extraction. China has designed two types TBMs to put in a whole test port, which are the helium cooled solid breeder (HCSB) TBM and the dual functional lithium lead (DFLL) TBM. A set of common tritium processing systems were designed for the two types of TBMs including the tritium extraction system (TES), the helium coolant system and the coolant purification system (CPS). Tritium in the TBMs and the ancillary systems should be controlled for the sake of the radiological safety. The design of some key parameters such as the partial pressures of tritium in the coolant and above the breeder, the flux of carrier gas for tritium extraction, the efficiency of tritium extraction, the selection of the structure material together with the surface coating, etc, depend on the structure of the TBMs and the control of tritium release to the environment. A set of calculation models based on the tritium mass balance among the TBMs and different ancillary systems were developed to appraise the tritium safety on the common tritium systems and the Chinese TBMs. Tritium permeation barriers coating on the structure material of some components and a double-wall design for some tritium containers, and a glove box atmosphere detritiation system (GDS) establishment are all necessary for the control of tritium release to the environment.

PS2-1302

TRITIUM PROCESSING SYSTEMS FOR HCPB-TBM

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One of the most challenging issues in the TBMs testing campaign to be accomplished in ITER is the correct and efficient recovery of the generated tritium.
For the European HCPB-TBM the main auxiliary systems involved in this task and directly interfaced with the TBM itself are the TES (Tritium Extraction System), which purpose is to extract from the ceramic breeder all the generated tritium, sending it to TEP (Tokamak Exhaust Purification) system, and CPS (Coolant Purification System) which extracts the permeated tritium from He-coolant and keeps controlled the chemistry of the primary cooling circuit.
Both TES and CPS have to be compatible with:
- the performance requirements inherent to the different ITER operational phases;
- the ITER space requirements;
- the requirements in terms of interface with the ITER tritium processing;
- the need to produce results "DEMO" or, more in general, "Reactor" relevant

In this paper, first of all the expected composition of the feed flow-rate to be processed by TES and CPS is given for all the foreseen ITER operational phases.
Moreover, the principal process options potentially able to fulfil the above mentioned requirements for both TES and CPS are indicated and discussed.
Finally, in the light of the operative conditions planned for the experimental campaign on TBMs, the most suitable process option for TES and CPS are presented together with their preliminary design, taking also into account the possibility to adapt these systems and the related technologies to the EU HCLL-TBM.

FAILURE MODE AND EFFECT ANALYSIS FOR THE EUROPEAN TEST BLANKET MODULES

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A Failure Mode and Effect Analysis (FMEA) at component level was done to study safety relevant implications arising from possible failures in performing Remote Handling (RH) operations.
Autonomous air cushion transporter, pallet, sealed casks and tractor movers needed for port plug mounting/dismantling operation were analysed. For each sub-system, the breakdown of significant components was outlined and, for each component, possible failure modes have been investigated pointing out possible causes, possible actions to prevent the causes, consequences and actions to prevent or mitigate consequences.
Off-normal events which may result in hazardous consequences for the public and the environment have been defined as Postulated Initiating Events (PIEs). Two safety-relevant PIEs have been defined by assessing elementary failures related to the analysed system. Each PIE has been discussed in order to qualitatively identify accident sequences arising from each of them. The two PIEs are:
• RHP Radioactive products (fraction of Dust & T implanted in VV) into Port Cell during RH operations for breach in “VV + cask” isolating boundary.
• RHG Cask stop and radioactive products (fraction of Dust & T implanted in VV) release into Gallery due to Cask leakage during transportation to Hot Cell.
For both PIEs radioactive release to the environment should not be a concern, according a first evaluation done in previous study. Nevertheless, further deterministic analysis could be required to determine response of safety systems (e.g.: efficiency of ventilation systems, isolation of HVAC) and effectiveness of rescue operations in mitigating the consequences and risks for workers. Precisely, even if the two PIEs do not lead to significant radioactive release to the environment, spreading of contamination inside the building and the operating areas can be induced. Consequently, for maintenance and/or decontamination activities, over radiation exposure to workers can be induced. Furthermore, fire hazard assessment should be required to demonstrate compliance of design features with safety limits also in case of fire triggered on board of the Transporter.
As an output of this FMEA study, also possible incidental scenarios, where intervention of rescue RH equipments is required to overcome critical situations determine by fault of RH components, were defined and grouped in seven families. Being rescue scenarios of main concern for Remote Handling activities, such families could be helpful in defining the design requirements of port handling systems in general and on transfer cask in particular. Furthermore, they could be useful in defining casks and vehicles to be used for rescue activities.

PS2-1403

HELIUM COOLED TEST BLANKET MODULE BOX BEHAVIOUR UNDER ACCIDENTAL PRESSURISATION.

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The Helium Cooled Lithium-Lead (HCLL) breeder blanket concept is one of the two breeder blanket lines presently developed by the EU for DEMO reactor. In the short-term so-called DEMO relevant Test Blanket Modules (TBMs) of these breeder blanket concepts shall be designed, manufactured, tested, installed, commissioned and operated in ITER for first tests in a fusion environment.
For the purpose of licensing such test module in the ITER facility, a safety assessment of the different possible accidental conditions has to be performed. This paper presents the results of the thermo-mechanical calculations that have been obtained in case of internal module leak occurring between the 8.0 MPa pressurised helium circuit and the lithium-lead circuit. In a conservative manner, it is also assumed that helium cooling of the TBM box is stopped at the accident initiation.
In such situation the TBM box will be entirely pressurised at 8.0 MPa and the different challenges are related to the thermo-mechanical behaviour, the plasma power hold up and the heat removal capacity of the module.
An assessment to estimate the time when the TBM box can withstand the different loads induced by the situation is presented. Particularly the available time span to trigger an emergency plasma shutdown is estimated in regards of the maximal allowable stress for the EUROFER which is the test blanket module material.
A 3D finite element model has been developed with CAST3M computer code taking into account the beryllium layer, the decay heat after shutdown and the thermal radiation phenomena. After a description of the model, the paper presents and explains the results and particularly the methodology followed to determine the maximal allowable stress location which must combine the thermal transient calculations and the pure mechanical calculation under the 8.0 MPa loading.
The results show that the hottest spot of the First Wall (FW) was the most challenged location and that the structure can withstand such accidental conditions without plasma shutdown up to 15 seconds. Afterwards, the Structural Design Criteria defined by ITER for In-vessel Components (SDC-IC) is not any more fulfilled. This does not necessary mean that a Lithium-Lead leakage can occur in the vacuum vessel. More research and development are needed to have a clear understanding of crack propagation and break size in a material like EUROFER where few data are available, particularly at the temperature reached in this accident.
Finally as a conclusion of the studies presented in this paper, it can be stated that, although very conservative assumptions were taken, the time span is large enough to trigger a plasma shutdown in order to avoid more severe operational and safety consequences of this accidental situation.

ACTIVATION AND AFTERHEAT ANALYSES FOR THE HCPB TEST BLANKET

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The Helium-Cooled Pebble Bed (HCPB) blanket is one of two breeder blanket concepts developed in the framework of the European Fusion Technology Programme for performance tests in ITER. The recent development programme focussed on the detailed engineering design of the Test Blanket Module (TBM) and associated systems including the assessment of safety and licensing related issues with the objective to prepare for a preliminary Safety Report.

To provide a sound data basis for the safety analyses of the HCPB TBM system in ITER, the afterheat and activity inventories were assessed making use of a code system that allows performing 3D activation calculations by linking the Monte Carlo transport code MCNP and the fusion inventory code FISPACT through an appropriate interface. A suitable MCNP model of a 20 degree ITER torus sector with an integrated TBM of the HCPB PI (Plant Integration) type in the horizontal test blanket port was developed and adapted to the requirements for coupled 3D neutron transport and activation calculations.

Two different irradiation scenarios were considered in the coupled 3D neutron transport and activation calculations. The first one is representative for the TBM irradiation in ITER with a total of 9000 neutron pulses over a three (calendar) years period. It was simulated by a continuous irradiation for 3 years minus the last month and a discontinuous irradiation with 250 pulses (420 s pulse length, 1200 s power-off in between) over the last month. The second (conservative) irradiation scenario assumes an extended irradiation time over the full anticipated lifetime of ITER according to the M-DRG-1 irradiation scenario with a total first wall fluence of 0.3 MWa/m2.

For both irradiation scenarios the radioactivity inventories, the afterheat and the contact gamma dose were calculated as function of the decay time. Data were processed for the total activity and afterheat of the TBM, its constituting components and materials including their breakdown into dominant radio-nuclides at each decay time.

This paper describes the main features of the coupled 3D neutron transport and activation calculations, presents the major results obtained for the two irradiation scenarios and discusses their importance for the safety analyses and the waste categorisation.

PS2-1405

RECENT PROGRESS IN SAFETY ASSESSMENTS OF JAPANESE WATER COOLED SOLID BREEDER TEST BLANKET MODULE

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Water Cooled Solid Breeder Test Blanket Module (WCSB TBM) is being designed by JAEA for the primary candidate TBM of Japan, and the safety evaluation of WCSB TBM has been performed. This reports presents summary of safety evaluation activities of the Japanese WCSB TBM, including nuclear analysis, source of RI, waste evaluation, occupational radiolysis exposure (ORE), failure mode effect analysis (FMEA) and postulated initiating event (PIE).

For the purpose of basic evaluation of source terms on nuclear heating and radioactivity generation, two-dimensional nuclear analysis has been carried out. By the nuclear analysis, distributions of neutron flux, tritium breeding ratio (TBR), nuclear heat, decay heat and induced activity are calculated. Tritium production is calculated by the nuclear analysis by integrating distributions of TBR values, as about 0.2 g-T/FPD.

With respect to the radioactive waste, the induced activity of the irradiated TBM is estimated, and compared with the DEMO blanket that is operated to 10MWa/m2 at the surface of the first wall. The induced activity of the TBM is one order of magnitude lower than the DEMO blanket.

For the purpose of occupational radiolysis exposure (ORE), RI inventory is estimated. Tritium inventory in pebble bed of TBM is about 3x10^12 Bq, and tritium in purge gas is about 3x10^11 Bq. Tritium is thought to permeate into cooling system from pebble bed and from first wall. In addition to tritium, active corrosion product (ACP) is thought to be generated in the cooling system.

FMEA has been carried out to identify the PIEs that need safety evaluation. PIEs are summarized into three groups, i.e., heating, pressurization and release of RI. PIEs of local heating are converged without any special cares. With respect to heating of whole module, two PIEs are selected as the most severe events, i.e., loss of cooling of TBM during plasma operation and ingress of coolant into TBM during plasma operation. Detail of heating of whole module is described in another report submitted on this symposium. With respect to PIEs about pressurization, the PIEs of pressurization of the compartment nearby the pipes of cooling system are evaluated, because rupture of the pipes result pressurization of such compartments, i.e., box structure of TBM, purge gas loop, TRS, VV, port cell and TCWS vault. Box structure of TBM is designed to withstand the maximum pressure of the cooling system. At other compartments, discharged coolant is released by mitigation systems and pressure does not exceed the design limit. With respect to PIEs about release of RI, there are three inventories of RI, i.e., RI in VV (tritium and radio-activated dust), RI in purge gas (tritium) and RI in coolant (tritium and Active Corrosion Products (ACP)). The sequences of release of these inventories of RI are evaluated.

It was concluded that the baseline data on nuclear heating, decay heat and generation of tritium and induced activity were clarified for further evaluation of ORE. Also FMEA was carried out to identify the important PIEs to be considered in safety analysis. By the safety evaluation activities, the basis of detailed safety assessment of the WCSB TBM was established.

PRELIMINARY SAFETY ANALYSIS OF KOREA HELIUM COOLED SOLID BREEDER TEST BLANKET MODULE

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ITER Test Blanket Module (TBM) which will be the act-alike module corresponding to DEMO blanket is aimed to verify the capability of tritium self-sufficiency and the extraction of high grade heat using tritium breeding module concepts for ITER missions. Conceptual design of Korea Helium-Cooled Solid Breeder (HCSB) TBM, one of two concepts that Korea has proposed, has been performed including performance analyses. In this paper, the results of preliminary safety analyses of Korea HCSB TBM are described. Loss of coolant into Vacuum Vessel, loss of coolant into breeding zone and ex-vessel loss of coolant are selected as three reference accidental scenarios for the TBM. A 2-D finite element model is established to examine the maximum temperature reached, the time and duration of the peak. A systematic assessment of the design is executed to investigate key safety functions such as activation, decay heat, waste disposal, radiological and energy source terms. It proves the robustness of the design from the safety perspective.

DETERMINISTIC SAFETY ANALYSIS OF THE REFERENCE ACCIDENTAL SEQUENCE FOR THE EUROPEAN HCPB TBM SYSTEM

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The European Helium Cooled Pebble Bed Test Blanket Module (HCPB TBM) is one of the blanket concepts studied in EU as possible candidate of DEMO blanket and for testing in ITER. The TBM is put in a horizontal port of ITER facing plasma burn directly with an own first wall (FW) of 1m². The TBM is cooled by Helium Coolant System (HCS) and is connected to auxiliary systems for T extraction/recovery. Failure mode and effect analysis (FMEA) on HCPB TBM has been performed in order to identify the possible postulated initiating events (PIE) and the most dangerous accidental sequences. Any failure of components in HCS affects the heat removal capability and TBM box with potential failure of the first containment vacuum vessel (VV). 4 PIEs have been considered, judged to cover all the most demanding accidents: 1) Loss of flow accident (LOFA), 2) Loss of coolant accident (LOCA) in ex-vessel, 3) Failure of HX (He/water), and 4) In-box LOCA with pressurization of the purge lines. A particular sequence in point 2) (ex-vessel LOCA with failure of Fusion Power Shutdown System (FPSS)) has been selected for the deterministic analysis in frame of EFDA licensing task. This paper presents the study of this sequence including assumptions and modeling.

The analysis regarding the accident evolution is divided in 3 phases. In phase1 "He blow-down" it is assumed that during the plasma burn a failure of the He coolant confinement in the tokamak cooling water system (TCWS) vault occurs, which means a double-ended pipe break in a large diameter pipe. TBM cooling is lost in short time. In phase 2 "Delayed plasma shutdown" it is assumed that the detection of ex-vessel LOCA fails to trigger FPSS. Be-cover facing plasma and TBM box are heated up by perpetual plasma burn (270 KW/m²), until EUROFER melting point (~1539°C) on FW surface or Be melting point (~1290°C) on Be-cover is reached. At this time a complete failure of the FW channels integrity is assumed with penetration of air in the VV and plasma disruption implies plasma shutdown. Using an ANSYS-model thermal analysis is done to determine the time of plasma shutdown and the level of temperature for possible damages in the TBM structure. In phase 3 "Long term behavior" the sequence continues with decay heat in the TBM and air ingress in the VV causing chemical reaction with Be-cover. In addition the failure of water coolant confinement in the VV allows steam ingress into the VV and its reaction with Be-cover releases heat and H_2 production. Air-steam mixture in the VV can enter into TBM box through damaged TBM structure and reacts with Be pebbles. Using same ANSYS-model as in phase 2, temperature transient for long term can be done and H_2 production can be estimated. Dust and activation products are transported from the VV throughout the damaged FW, the piping to the double-ended break in the TCWS vault.

Based on the results for the accident evolution appropriate safety method against accident consequence can be arranged.

PS2-1409

ASSESSMENT OF THE ACTIVATION, DECAY HEAT, AND WASTE DISPOSAL OF THE US HELIUM-COOLED CERAMIC BREEDER TEST BLANKET MODULE IN ITER

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The radioactivity inventory and after heat in the U.S. helium-cooled ceramic breeder (HCCB) test blanket module (TBM) have been accessed at shut down and for several times thereafter. Also assessed is the waste disposal rating (WDR) of its various components. The objectives are: (1) to provide the information needed for further safety assessment of the generated radionuclides and their volatility, as well as after heat on the safety operation of ITER, and (2) to aid in determining the waiting cooling period prior to removing and transporting the TBM for further treatment outside ITER site. The TBM is proposed to be placed in one of the three dedicated test ports of ITER. The current proposal is that it will occupy 1/3 of the horizontal upper half of a port next to Japan and Korea sub-modules. The sub-module will have its own FW and structural container box that houses the breeder and beryllium pebble bed units, arranged in an edge-on-configuration. Helium is used to cool the FW, sides of the box, and the internal plates. Conventional ferritic steel (F82H) is used as the structure. The sub-module has 71 cm height, 38.9 cm wide and 60 cm depth in the radial direction. The breeder beds are made of Li2TiO3 pebbles with 94% theoretical density and 62% packing factor (as the beryllium pebbles). Lithium-6 is enriched to 75%. A 2 mm thick beryllium layer is used as a plasma facing material on the FW area subjected to 0.78 MW/m2 neutron wall load. Pulsed operation mode is assumed. Each pulse is assumed to be 400 s full flat top followed by 1800 s dwell time, during which the decay of the generated radionuclides are accounted for. The 500 MW pulses are assumed to be generated one after another until a fluence limit of 0.3 MWa/m2 is reached without replacing the TBM. This gives upper conservative estimates for the radioactive inventory and decay heat. During operation in the D-T phase, the total heating rate in the TBM is ~263 KW. The total amount of tritium generated in the breeder and the beryllium multiplier is ~9 g and 0.07 g, respectively, after reaching the 0.3 MWa/m2 fluence limit. At shutdown, the total radioactivity and after heat levels are ~0.89 MCi and ~0.002 MW respectively. These values drop sharply after one minute to ~0.098 MCi and ~0.0006 MW. The contribution from the F82H structure is the dominant up to ~10 years following shutdown. In this time frame, the activation levels in the breeder and beryllium are lower than those attained in the structure by ~2 and ~6 orders of magnitude, respectively (~1 and ~7 orders for after heat). After ~10 years, the contribution to the total activation and after heat from the breeder is the dominant one due to the generated tritium. The contribution from the first wall and its coolant channel and from the first breeding zone (10 cm-thick) is the dominant. The WDR of various components are far below unity and thus the impact on safety and waste disposal is minimal and well within ITER regulatory guidelines.

LIQUID BLANKET MHD EFFECTS EXPERIMENTAL RESULTS FROM LMEL FACILITY AT SWIP

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The self -cooled /helium-cooled liquid metal blanket concept is an attractive ITER and DEMO blanket candidate as it has low operating pressure, simplicity, and a convenient tritium breeding cycle. But MHD pressure drop remains a key issue, especially in ducts with flow channel inserts (FCI), where the reduction in MHD pressure drop is difficult to predict with existing tools, and there are no available experimental data to check current predictions. To understand well various kinds of MHD effects, it is important for us to analyze and understand FCI effects. In this paper, we present measurements of the MHD effects due to off normal power shutdown, two¨Cdimensional effects due to channel velocity profiles, three-dimensional effects caused by manifolds, and surface/bulk instability effects as a result of insulator coating imperfections. These results were collected from the Liquid Metal Experimental Loop (LMEL) facility at Southwestern Institute of Physics, China and in collaboration with Argonne National Laboratory, US under an umbrella of the People¡¯s Republic of China/United States program of cooperation in magnetic fusion. Some results were observed for the first time, such as two dimensional effects and instabilities due to insulator coating imperfections. The experiments were conducted under the following conditions: a uniform magnetic field volume of 80x170x740mm and a maximum value of magnetic field, B0, of 2 Tesla. The mean flow velocity V0 was measured with an electromagnetic (EM) flow meter (error of 1.2%); a Liquid-metal Electro-magnetic Velocity Instrument (LEVI) was provided by Argonne National Laboratory. The flow was driven by two Electro-magnetic (EM) pumps (6.5 +11.6 m3/h); the operating temperature was 85 centigrade degee due to self-heating by the EM pump and friction of the fluid against the loop piping. Experimental parameters were: Hartmann number, M, up to 3500, velocity v0 up to 1.2 m/s under magnetic field, and B0=1.95 Tesla. Analysis of the FCI results, using simplified modeling based on the experimental results, are also included. In this paper, we also review the progress made by SWIP in this research field.

MHD/HEAT TRANSFER CONSIDERATIONS FOR THE DCLL BLANKET FOR DEMO AND ITER TBM

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The Dual-Coolant Lead-Lithium (DCLL) blanket is considered in the US for testing in ITER and as a primary candidate for a DEMO reactor. In this blanket, eutectic alloy PbLi circulates as a breeder and coolant, while He is used for cooling the reduced activation ferritic steel structure. The key element of the DCLL concept is a flow channel insert (FCI) made of a silicon carbide composite (SiCf/SiC) or possibly SiC foam, which is used to minimize a heat leakage from the breeder zone, to separate hot PbLi from the ferritic structure, and to reduce the MHD pressure drop. The thermal performance of the blanket is strongly affected by magnetohydrodynamic (MHD) phenomena. Here, we summarize results for MHD flows and heat transfer in a unit cell (front poloidal channel) of the DCLL blanket for three scenarios: DEMO, ITER H-H, and ITER D-T. The most important MHD phenomena that may affect heat transfer in the PbLi include formation of the near-wall jets, buoyancy-driven flows and two-dimensional MHD turbulence. In the DEMO scenario, preliminary FCI parameters have been identified via numerical simulations to meet basic safety and operational requirements. The performance of the FCI depends strongly on the flow in the central channel and the gap, which in turn depend on the electrical conductivity of the FCI and the flow regime. Both laminar and turbulent flow conditions have been considered. The most critical requirement is that on the maximum temperature difference across the FCI, which may be too high, leading to intolerably high thermal stresses in the FCI. In the ITER D-T scenario, both surface and volumetric heating are present. The exit breeder temperature is limited to 470C. In the ITER H-H scenario, no volumetric heating is included, but instead the LM is assumed to enter the module at 470, while the helium inlet temperature is at 300C. This allows studying the heat transfer from the liquid metal to the helium flow. For all scenarios, we have evaluated heat losses into the helium streams and calculated temperature distributions in the FCI. Both ITER scenarios in normal conditions look to be acceptable, i.e. all restrictions on the interface temperature, thermal stress in the insert, etc. can be easily met. For the ITER D-T scenario we have also analyzed heat transfer in off-normal conditions, when the flow in one of the channels is reduced or fully stopped. In the concluding section, we discuss the testing approach in ITER (including support from modeling and experiments) aiming at data which can be extrapolated to the DEMO conditions.

RECENT DEVELOPMENTS IN NEUTRONICS

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The broad approach to fusion presents a number of new changes for neutronics calculations which must cover the analysis of ITER, DEMO, spherical tokamaks and IFMIF. Radiation transport modelling is essential for both the design of these machines and as input to licensing. Recent developments in the field of neutronics are characterised by the need for greater detail, a more rapid production and dissemination of results and the application of new computer technology to address these issues.

In recent years Monte-Carlo codes such as MCNP and McBend have been deployed on parallel machines to allow larger more detailed calculations to be carried out. The latest discrete ordinates codes such as Attila can model detailed geometries in 3D. These developments permit more sophisticated representations of the components and the ability to derive their specification directly from CAD designs is being developed.

The benefits of these developments are demonstrated in this paper using examples of the design of components for ITER and of IFMIF. An analysis of the RF antenna using Attila was used to assist in the design of shielding in order to minimise the activation behind the port plug; a study of the LIDAR and the polarimeter system was used to aid port integration; and a comparison the efficiency of discrete ordinates codes with Monte-Carlo methods was carried out using calculations of the shielding around the IFMIF target. The use of complementary techniques allows the solution of previously intractable problems related to deep shielding and solution accuracy.

These examples also serve to illustrate some of the difficulties and advantages of the use of computer aided design which have implications for the wider design philosophy of ITER.

The problem of the dissemination of the results to design engineers and the archiving of results if is being addressed by the development of an integrated results database and QA system.

EXPERIMENTAL STUDY OF MHD EFFECTS ON HEAT TRANSFER CHARACTERISTICS ON TURBULENT PIPE FLOW OF FLIBE SIMULANT FLUID

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This paper presents experimental results on MHD turbulent heat transfer in a circular pipe using Flibe simulant fluid. Flibe is considered as a promising candidate for coolant and tritium breeder in some fusion reactor design concepts because of its low electrical conductivity compared to liquid metals. This reduces the MHD pressure drop to a negligible level; however, turbulence can be significantly suppressed by MHD effects in fusion reactor magnetic field conditions. Heat transfer to the Flibe coolant is characterized by its high Prandtl number. In order to achieve sufficient heat transfer and to prevent localized heat concentration in Prandtl number fluid coolant, high turbulence is essential. Even though accurate prediction of the MHD effects on heat transfer for high Prandtl number fluids in the fusion environment is very important, reliable data is not availalbe especially for conducting wall ducts. Therefore, it is important to investigate MHD effects on heat transfer characteristics for high Prandtl number fluids in conducting wall pipe. In the current experiments, an aqueous solution of potassium hydroxide is used as a simulant fluid of Flibe, which is an electrically conducting high Prandtl number fluid. A series of experiments is performed by using FLIHY loop at UCLA established under JUPITER-II program. The small diameter stainless steel pipe with 50mm diameter is selected in order to reduce buoyancy effects and to achieve longer entrance length. These test sections are placed under maximum 2 Tesla uniform magnetic field for 1.4m in the axial distance. The radial temperature distribution of the fluid flow in the pipe is measured by means of thermocouples tower, which is consisted of inconel sheathed K-type thermocouples with a diameter of 0.13mm arranged from the inner wall surface to the center of the pipe. Measurable minimum distance from the inner pipe wall is 0.05mm. By authors•f previous work, it was found that degradation of Nusselt number is deviated from the pre-established experimental formula with increasing interaction parameter and suggested that the balance of laminarization due to strong magnetic field and induced thermal stratification is changed around the value of interaction parameter. Present study is investigated the effect of both vertical and lateral magnetic field on the temperature filed and aims to clear the interaction between laminarization and thermal stratification. The experimental results show that thermal stratification is generated at only small temperature difference between top and bottom of the pipe cross-section under magnetic field. At high Hartmann number, the region of thermal stratification is expanded and the temperature field is observed as almost fully laminarization, that is, the degradation of Nusslet number is promoted.

HELOKA FACILITY: THERMO-HYDRODYNAMIC MODEL AND CONTROL

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This paper presents the thermo-hydrodynamic model used to simulate the behaviour of the HELOKA (Helium Loop Karlsruhe) facility and describes the mechanism used to control various loop parameters. This test facility, which is under construction at the Forschungszentrum Karlsruhe (FZK), is designed for testing of various components for nuclear fusion such as the Helium-Cooled Pebble Bed blanket (HCPB) and the helium-cooled-divertor for the DEMO power reactor. Besides the individual testing of the blanket and divertor modules, the understanding of the behaviour of their cooling systems in conditions relevant for ITER operation is mandatory. An important aspect in the operation of these cooling loops is the accurate control, via feedback, of the flow parameters at the inlet of the test module. Understanding heat transfer and fluid flow phenomena during normal and transient operation of HELOKA is essential to ensure the adequacy of safety features. Systems analysis codes, such as RELAP5- 3D, are suited to this task. However, the application of these models to HELOKA design must be later validated by experimental measurements, while the basic physical models have been proven for light water reactors.
The control of the test section inlet parameters is one of the most important issues. In particular, the start-up phase, when the test section temperature is increased from ambient temperature up to 300°C, requires special attention. As a first step, the HELOKA open loop thermal transient was computed using the RELAP model. The data obtained have been used for the identification of the power-temperature transfer function needed to compute the parameters of the feedback controller (PID) using MATLAB and SIMULINK. An accurate control of the temperature during the start-up and flat top phases is achieved solely by controlling the heater power. The adopted solution reduces the harmonic distortions when operating at reduced power while keeping the investment cost low. This control has been implemented back into the RELAP model of HELOKA. Qualitatively, RELAP5-3D's predictions agree closely with those of the other system codes. Quantitatively, RELAP5 - 3D computes slightly higher temperature oscillations at the inlet of the TBM than the other system analysis.

CONTAMINATION OF ITER CORE BY HIGH-Z IMPURITIES AFTER ELMS

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The H-mode confinement in the future tokamak ITER is anticipated to be repetitively worsened by edge localized instabilities (ELMs). At each ELM the deuterium-tritium (DT) plasma lost from the pedestal into the scrape-off layer (SOL) and further onto a divertor armour produces surface erosion. The following contamination of SOL by the eroded and then ionized material species provides the impurities in the pedestal and the core. The fraction of carbon-based material (CBM) must be minimized, because of not acceptable accumulation rate of radioactive tritium inside its bulk. Therefore tungsten-based material (WBM) should be preferable even near the separatrix strike point (SSP) where DT-plasma maximum flux impacts on CBM tiles. However, the presence of highly but not fully ionized W-ions in the confinement region may get dangerous for the device operation, which is due to enhanced heat loss by the line radiation of W-ions.

In this work the DT-plasma contamination after the Type I ELMs is simulated with the tokamak integrated modelling code TOKES. The simulations imply some ELM-caused heat flux distribution over the wall as a function of time and poloidal coordinate along the divertor surface. The processes of emission of eroded C- and W-atoms and their ionization in the SOL as well as the multi-fluid transport in the confinement region among D-, T-, He-, C- and W-ions are calculated for the whole ITER discharge with multiple ELMs, different fuelling and auxiliary heating schemes (neutral beams and pellets) and the burning at fusion gain $Q < 10$.

The aim is obtaining tolerable ELM energy based on the radiation losses and deterioration of fusion gain caused by the W-impurity that was produced after the ELM. The TOKES calculates self-consistently also both the poloidal field external coil currents and the confined plasma currents and thus the separatrix dynamics, and thus a significant broadening of power footprint compared to the usual assumption of the SOL effective width of ion gyro-
radius. This allowed self-consistent calculations of heat flux at SSP that varies in time, and thus the heat flux distribution in zero-limit of SOL width. The corresponding contamination of DT-plasma by the W-ions (and C-ions) is examined in its dependence on the poloidal position of close juncture of CBM and WBM surfaces.

FUSION-RELATED WORK AT THE NUCLEAR ENERGY AGENCY DATA BANK

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The OECD Nuclear Energy Agency (NEA) Data Bank is part of an international network of data centres in charge of the compilation and dissemination of basic nuclear reaction data. Through its activities in the reaction data field, the NEA participates in the preparation of data for the modelling of future nuclear facility concepts and the development of reactor installations. A working party at the NEA on international nuclear data evaluation co-operation (WPEC) is established to promote the exchange of nuclear data evaluations, measurements, nuclear model calculations and validation. WPEC provides a framework for co-operative activities, such as the high priority request list for experimental data of special interest for certain applications, such as IFMIF or ITER.

The NEA Data Bank administrates the collection and validation as well as the distribution of the Joint Evaluated Fusion and Fission (JEFF) library, where the activities in the European Fusion and Activation File projects (EFF and EAF respectively) play an important role for new data evaluations. The topics cover verification of activation and transport data, calculation methods and validation via integral experiments. The EFF project brings together all available expertise in Europe related to the nuclear data requirements of existing and future fusion devices, and the project contributed greatly to the internationally recognised nuclear data library JEFF-3.1, released in May 2005. The NEA also provides tools for the EFF project, such as computer codes for nuclear energy and radiation physics applications. Of special interest for fusion applications are the integral experiments collected in the Shielding Integral Benchmark Archive Database (SINBAD) database. SINBAD is an internationally established set of radiation shielding and dosimetry data containing over 80 experiments relevant for reactor and accelerator shielding. About 30 of these experiments are dedicated to fusion blanket neutronics.

Materials research is a field of growing relevance for innovative nuclear systems, such as Generation IV reactors, critical and sub-critical transmutation systems and fusion devices. The NEA is organising a workshop on Structural Materials for Innovative Nuclear Systems (SMINS) aiming at stimulating an exchange of information on current material research programmes for different innovative nuclear systems in order to identify and develop potential synergies.

In this paper an overview will be given of the fusion-related projects within the NEA, with examples of nuclear data services offered, such as the SINBAD database and validation of data with fusion neutronics shielding experiments. The main emphasis will be given to recent work within the EFF project as well as a discussion on the forth-coming evaluation efforts among the EFF collaborators and conclusions from the SMINS workshop.

COUPLED TRANSIENT THERMO FLUID- STRESS ANALYSIS APPROACH IN A VTBM SETTING

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A virtual test blanket module (VTBM) has been visualized as a utility to aid in a fast, streamlined and optimized TBM design effort by facilitating integrated multi-code multi-physics modeling activities. Within this effort, a systematic integrated design approach, through the coupling of different analysis fields, has been established for steady-state TBM design performance analysis. In such an approach, a complete CAD model of the TBM, including the solid and fluid components are input into a CFD system, which provides the temperature field in the solid and fluid parts as well as the complete 3D flow distribution of the coolant in the manifolds and the coolant channels. The temperature field in the structural components is imported as a loading condition into the structural analysis system. This includes a series of utilities provided along with the CFD system that enable data exchange and mesh interpolation across the CFD mesh and the structural analysis mesh. Furthermore, in the proposed analysis methodology, the volume mapping technique is chosen because it allows for the thermal flow calculations and thermal stress calculations on specialized tailor made meshes, thus allowing for increased flexibility and improved accuracy.

There are cumbersome executive procedures involved in the data transfer when a coupled thermo-fluid and thermal stress analysis is considered. Such troublesome operations include multi-steps coupling, change of element form low to high order, and interpolation of data from low to high order elements. On the other hand, since complex transient analysis tends to involve considerable computer CPU time, the calculation would become ineffective without adopting proper data transfer techniques that take into account that the thermal time constant of a thermal field is much shorter than that of a fluid field. In this paper, procedures involved in transient coupled analysis are investigated in order to establish a reasonable calculation method. The established procedures are applied to study the impact on the design from a transient phenomenon like a time varying heat flux on the first wall due to off normal events.

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QUALIFICATION OF TRITIUM PRODUCTION MEASUREMENT TECHNIQUES FOR THE HCLL-TBM NEUTRONICS EXPERIMENT

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Neutronics experiments with a mock-up of the EU Helium-cooled Lithium-Lead Test Blanket Module (HCLL-TBM) are scheduled for 2007 with the aim to validate nuclear design parameters. The HCLL-TBM mock-up will consist mainly of LiPb with 15.7 at-% Li. Tritium production rates (TPR) will be measured with the lithium carbonate method applying a wet chemistry procedure with liquid scintillation counting. However, preliminary calculations reveal that the tritium production density in deeper positions in the assembly will be less than the lower limit for the lithium carbonate method. A higher sensitivity can be expected if the tritium producing events are measured rather than the accumulated tritium activity. Therefore, LiF thermo-luminescence detectors (TLD) are investigated at TU Dresden for their applicability as TPR probes for the HCLL-TBM mock-up experiment. The TPR can be derived from the measured TLD signal if the dose deposited in the TLD can be separated into the contribution from the alpha particles and tritons from the tritium breeding reaction and contributions from photons and charged particles from other reactions. To qualify the method, irradiation experiments with a LiAl-Pb assembly (size 40cm x 40cm x 60cm) simulating the HCLL-TBM mock-up have been performed with the DT neutron generator of TU Dresden. LiF TLD with natural isotopic composition and enriched in Li7 were inserted into the assembly. The DT neutron source fluence was selected so that the local fluence at the detector position is similar to the lowest fluence expected in the HCLL-TBM mock-up during the neutronics experiment. To eliminate the dose from photons and other charged particle reactions, the difference of the TLD signals from the two types of LiF TLDs is used as a measure for the TPR from Li6. The signal was calibrated with a Li-glass scintillation detector placed at the position of the LiF detector in the center of the assembly. The TPR profile obtained from the LiF detectors placed perpendicular to the assembly's z-axis in steps of 4 cm was compared with MCNPX calculations of the TPR and agreement within less than 10% was found.

Optically stimulated BeO detectors and track detector foils are applied for additional quantification of contributions from photons and charged particle reactions other than the tritium breeding reaction. The analysis is underway.

DEVELOPMENT OF HELIUM-COOLED FUSION APPLICATIONS: OVERVIEW ON MAJOR HELIUM ACTIVITIES AT THE FORSCHUNGSZENTRUM KARLSRUHE

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Helium cooled high temperature components and reactors from today`s point of view are most likely to capture a major importance in the future energy market. Similar to the tendency in the development of future fission reactors the main emphasis in regard to future fusion reactors is on Helium cooled systems. Both European reference blanket concepts are completely Helium cooled and in addition, the development of a Helium cooled divertor is in progress. Also the third, alternative European blanket concept "Dual-coolant lead lithium" in regard to a DEMO fusion reactor relies on a Helium-cooled structure. In the International-Fusion-Material-Irradiation-Facility (IFMIF), again, Helium cooling is applied e.g. to the High Flux Test Module (HFTM). Against this background major Helium activities were launched at the Forschungszentrum Karlsruhe, including the design and construction of several Helium Test Facilities applicable to perform various experiments from single effect studies up to full component tests for the qualification of complete test modules to be operated in ITER. In addition a fundamental research Programme is under way to improve the local Helium cooling technologies applied in different applications and to improve the knowledge base on heat transfer, boundary layers, turbulence development and flow structures as well as the dynamic behaviour of large Helium cycles under unsteady boundary conditions. An important complementary activity is found in the area of computational fluid dynamics (CFD) were the most appropriate turbulence models are determined by code validation based on the obtained experimental data.

In this paper an outline of the overall Helium cooling development strategy at Forschungszentrum Karlsruhe is given along with a brief description of the operated and planned Helium Test Facilities including the (i) ITHEX facility used for flow field studies in IFMIF-relevant transitional flows, (ii) the HEBLO facility used for example to test Blanket First Wall channels and divertor mock-ups, (iii) the HELOKA-HP/TBM facility used for TBM development and (iv) the HELOKA-LP facility used for IFMIF HFTM development. In addition an overview on the status of research on jet impingement cooling applied to the divertor, rib cooling for the First Wall of breeding blankets and convective heat transfer in minichannels of the IFMIF HFTM is given.

DEVELOPMENT OF MULTI-SCATTERED TIME-OF-FLIGHT NEUTRON SPECTROMETER TO MEASURE DT FUEL RATIO IN FUSION EXPERIMENTAL REACTOR

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The DT burn control in magnetic confinement fusion devices requires real time information on the fuel ratio (the ratio of deuterium density and tritium density) in the plasma. The fuel ratio in a DT burning plasma can be derived from the intensity ratio of DD/DT neutrons, and detecting trace amount of DD neutrons in the DT burning plasma is a key issue. Neutron spectroscopy is superior to other tools in the diagnostics for plasma core where fusion reactions most frequently occur. Time-of-flight neutron spectrometer is a candidate for the fuel ratio measurement system in International Thermonuclear Experimental Reactor. We have been developing a new type of neutron spectrometer to monitor the fuel ratio in the core of the ITER plasma. The system is based on a conventional time-of-flight method and composed of a water cell as a neutron scattering material and a few tens of scintillator pairs distributed around the first scintillator in a corn shape, which we call a multi-scattered time-of-flight neutron spectrometer (MS-TOF). The intensity ratio of the DD/DT neutron is enhanced approximately three times before reaching the TOF crystals through elastic scattering with hydrogen nuclei in the water cell. The DD neutrons can be detected easier by enhancing their relative intensity, together with radiation tolerance of the detection system.
Here we mainly present a trial experiment for the prototype MS-TOF system by detecting trace-DD neutrons within a DT neutron beam (20-mm diameter) at the Fusion Neutronics Source (FNS), Japan Atomic Energy Agency. The FNS is an accelerator, which bombards a tritium-storage target with a deuterium beam to generate DT neutrons, and simultaneously produce a fraction of DD neutrons by the self-accumulation of D on the target. The cross section of DD reaction predicts that generation ratio of DD/DT neutrons is around 1% for the FNS neutron generator. The experimental results have shown that the DD and DT neutron peaks are clearly observed, and the measured intensity ratio of the DD/DT neutrons is about 1.9%, which would be reasonable for the tritium target that had much poorer tritium retention after an excessively prolonged operation. The trial experiment has successfully demonstrated the feasibility of the MS-TOF concept for detecting trace-DD neutrons within a DT neutron beam extracted from a DT burn plasma.

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* K. Asai, N. Naoi, T. Iguchi et al., Rev. Sci. Instrum. 77, 10E721 (2006).

INTERPRETATION OF LEVI VELOCITY SIGNALS IN 3D MHD FLOWS

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For experimental determination of velocities in magnetohydrodynamic liquid metal flows, probes that measure electric potential gradient are often used. They are known as Liquid-metal Electromagnetic Velocity Instrument (LEVI) and they have been used in the past preferentially for investigating more or less fully developed flows in poorly conducting duct and flows with smooth variations along the channel axis. For such applications, where electric current density is negligible, the probe gives reliable results since the potential gradient signals can be directly interpreted as a velocity measure.

If the flow varies along its path on very short length scales, like in ducts with abrupt change of cross section or in manifolds, 3D electric currents may occur that are not negligible any more so that the LEVI readings may become inaccurate. Moreover the presence of the probe itself may introduce already strong perturbations to the flow field since internal layers spread along magnetic field lines that touch the shaft of the probe at both sides. This effect perturbs the flow not only locally in the vicinity of the probe tips but along an entire region located between the internal layers. The present paper aims in quantifying these effects in order to obtain reliable velocity data.

AN EASY WAY TO PERFORM A RADIATION DAMAGE CALCULATION IN A COMPLICATED GEOMETRY

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With the advent of ITER the need for more and more accurate fusion neutronics analyses increases. The standard software package for these analyses is MCNP, by which 3-dimensional radiation transport analyses may be performed. Demands on requested parameters are expanding from the more traditional neutron flux values to all kinds of derived parameters. These include radiation heating and reaction rates. Also for components receiving a high neutron flux the calculation of radiation damage is an important issue. Usually approximations are made to calculate the radiation damage. In this paper it is shown that with negligible additional effort an almost exact analysis is possible in arbitrary complicated situations.

A correct estimation of the expected radiation damage of components involves the use of molecular dynamics simulations. However, this is only possible for scientific investigations due to the extremely long calculation times needed for actual engineering studies. Hence, one nearly always resorts to using pre-compiled damage cross-section data, which are used as response cross sections in the MCNP analyses. One often needs an evaluation of the radiation damage in a material (e.g. stainless steel). Therefore in the current approach one frequently uses pre-calculated damage cross-section data for a few specific materials in a relatively coarse energy group structure (e.g. the 640-group SAND-II structure such as used in the DAMSIG library).

In this approach the details of the analysis are only taken into account to a limited extent. Important factors can only be approximated. These include the composition of materials (is in the actual problem the material composition identical to the one on the library?) the effect of temperature (is the actual temperature identical to the one on the library?) and self-shielding (the use of group cross-section data always leads to approximations).

In the current approach continuous-energy damage cross-section data are used, which were generated by the nuclear-data processing code NJOY. The data are available to MCNP as response cross sections. A damage calculation in MCNP for a material mixture involves many isotopes, which makes the standard method in MCNP completely impracticable due to the large amount of of pre- and post-processing.

In this paper it is shown, that these continuous-energy damage cross-section data may be used in a much simpler way by using a modified material specification, which is weighted by the damage cross section of the isotopes.

A theoretical foundation for this approach is given. The data are compared with damage cross-section data from literature, showing a good agreement. It is demonstrated that, without additional effort, a calculation of radiation damage is possible which is completely consistent with the underlying radiation transport calculation. This greatly simplifies these calculations and enables a calculation of radiation damage as a standard deliverable in ITER analyses.

DATA COLLECTION ON COMPONENT MALFUNCTIONS AND FAILURES OF JET ICRH SYSTEM

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The objective of the activity was to collect and analyse data coming out from operating experiences gained in the Joint European Torus (JET) for the Ion Cyclotron Resonance Heating (ICRH) system in order to enrich the data collection on failures of components used in fusion facilities.
Alarms/Failures and malfunctions occurred in the years of operations from March 1996 to November 2005, including information on failure modes and, where possible, causes of the failures, have been identified. Beyond information on failures and alarms events, also data related to crowbar events have been collected. About 3400 events classified as alarms or failures related to specific components or sub-systems were identified by analysing the 25 hand-written logbooks made available by the ICRH operation staff. Information about the JET pulses in which the ICRH
system was operated has been extracted from the tick sheets covering the whole considered time interval. 20 hand written tick sheets cover the period from March 1996 to middle May 2003, while tick sheets recorded as excel files cover the period from May 2003 to November 2005. By analysing the tick sheets it results that the ICRH was operated during about 12000 plasma pulses.
Main statistical values, such as rates of alarms/failures and corresponding standard errors and confidence intervals, have been estimated. Failure rates of systems and components have been evaluated both with regard to the ICRH operation pulses and operating days (days in which at least one ICRH module was requested to operate). Failure probabilities on demand have been evaluated with regard to number of pulses operated.
Some of the results are the following:
• The highest number of alarms/failures (1243) appears to be related to Erratic /No-output of the Instrumentation and Control (I&C) apparatus, followed by faults (829) of the Tetrode circuits, by faults (466) of the High Voltage Power Supply system and by faults (428) of the Tuning elements.
• The maximum number of events related to I&C (595) led to anomalous operations of CODAS, followed by 125 anomalous operations of stubs.
• The total number of operation pulses for the four ICRH modules is of 44216; that corresponds to a total (integrated for the four modules) of 5280 days of pulse operation.
• The number of failures/alarms of the ICRH system increases quite linearly with the number of pulses in which the system is operated.
• A crowbar event happened on average every 9 ICRH pulses.
• The rate of failure on demand of ICRH module is of about 0.10/pulse.

EX-SITU TRITIUM REMOVAL FROM JET TILES USING RF INDUCTIVE HEATING

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Radio-frequency (RF) inductive heating was successfully used as an ex-situ technique for the detritiation of entire tiles retrieved from JET. The assessment of the detritiation process is performed using three different techniques namely, autoradiography, calorimetry and full combustion asisted by the liquid scintillation analysis. Autoradiography showed that using the RF technique more than 99% of the total tritium inventory can be efficiently removed from a tile after several heating cycles at the average temperature of only 490°C. On the other hand, a comparison of the combustion measurements obtained before and after RF heating, has shown that the bulk tritium activity decreased significantly (95% of the bulk tritium was released) while at the same time more than 99% of the surface tritium is liberated.
Nevertheless, the Decontamination Factor (DF) achieved by the RF treatment is not sufficient in order to qualify the method (as such) as a potential ex-situ detritiation technique. To reach the objective of LLW category waste, the tritium residual activity on the tile should not exceed the 12 kBq g-1. However, it should be mentioned the tile was heated at relatively low temperatures (maximum 490°) which is not enough to efficiently release the bulk tritium.
The three methods used to assess the RF detritiation process have their pros and cons i.e. calorimetry allows a rough estimation of the complete tile, IP imaging allows an approximate estimation of the tritium content before and after detritiation but only for the surface tritium, while full combustion associated to the scintillation analysis allows very accurate measurements but only for specific samples having reduced dimensions therefore, extrapolation is needed to get an estimation of the tritium content of a entire tile. The detritiation efficiency of each method obtained for the MKIIA JET divertor tile BN4 is reported and compared.

PS2-3004

DESIGN STUDY OF A FIRST WALL IN JT-60SA FOR REMOTE HANDLING MAINTENANCE

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JT-60 is planned to be modified as a fully superconducting coil tokamak (JT-60 Super Advanced, JT-60SA) to establish scientific and technological bases for an economically and environmentally attractive DEMO reactor. It will be also a satellite tokamak in a part of broader approach for ITER. It is designed for high normalized beta and steady-state research in a break-even class DD plasma for 100s or longer. Plasma heating power of 41-50 MW with duration of 100s will be planned for high confinement plasma research at high density relevant to ITER. The heating power less than 15MW is considered as an extended future option for day-long operating demonstration. The remote handling capability for in-vessel components should be required due to the radio-activation of a vacuum vessel.

Expected heat fluxes on a first wall with assuming local asymmetry of twice are 0.3 MW/m2 during 100s for highly radiative plasma with high power heating and 0.05 MW/m2 for day-long operation. An armor tile bolted on a water cooled heat-sink is applied to exchange each armor tile for repairing damaged armor and research on plasma material interaction. Heat transfer rate more than 0.1 MW/m2 between a tile and a heat-sink is required to keep temperature of a fixing bolt less than 600 deg-C. A water-cooled heat-sink made of stainless-steel will be applied for a normal first wall.

Each armor tile has holes for one fixing bolt and gripping for remote handling manipulator. Poloidal and toroidal key structure is introduced between heat-sink and armor tile to position and support against rotational force on a tile. A fixing bolt has spring structure to keep enough fastening for thermal contact between a tile and a heat-sink at cyclic electromagnetic forces and thermal stress. Thick carbon and thin metal armor tiles should be able to be fixed by the same structure for plasma wall interaction research in conventional low Z wall and DEMO relevant metal wall with exchange by remote handling. Total height of the inner first wall is limited at 70 mm to allow low aspect plasma for high beta plasma research, which includes thickness of an armor tile and a heat-sink and minimum gap of 15 mm for magnetic sensors, tube for pellet guiding and boronization between a heat-sink and a vacuum vessel. Therefore, compact and simple structure is required for tile fixing structure.

Some parts of an inner first wall act as the NB armor for shine-though of positive ion sourced neutral beam, where expected maximum heat flux is 2 MW/m2 during 100s. A water cooled copper alloy heat-sink and improvement of thermal contact between a tile and a heat-sink are required for heat removal. Thin graphite foil between a tile and a heat-sink is effective for thermal contact, but the handling by manipulator is important issues for remote handling and first wall design.

CONCEPTUAL DESIGN OF JT-60SA CRYOSTAT

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JT-60U modification program to fully superconducting device has been proceeded, namely "JT-60SA", toward early realization of fusion energy based on tokamak concept.
The design of JT-60SA cryostat is expected to achieve a vacuum thermal insulation for super conducting coils, a bio-shielding boundary and structural gravity support. The cryostat is required to cover JT-60SA tokamak device, which is 15 m of total height and 7 m of radius, but there is geometrical limit due to surrounding devices reutilized. Although the cryostat consists of vessel body and gravity support, and the structural material is low cobalt 304 stainless steel (Co: < 0.05 wt%).
The vessel body consists of 9 parts split, which are a double-walled structure filled with boron-doped shielding concrete. The shape of cryostat is specified with these utilizations of ports since vacuum vessel and surrounding devices are jointed together with bellows and electrical insulator at the cryostat. These structural conditions result in the spherical shape of the vessel body and adjust to 18 sectors of the JT-60SA. Basically the operational load is atmospheric pressure so that vessel body parts are bolt-jointed and lip-sealed at each flange.
Gravity support structure is composed of 9 legs and two wider ring structures, which are connected each other and support vacuum vessel and magnets. A pedestal base plate is bolted on the torus hall by 80 bolts (M64). The operational loads of gravity support are total device gravity of 2500 ton, and electromagnetic force of the vacuum vessel and superconducting magnets.
Inner surface of the cryostat is covered with 80K thermal shield, which is made of 304L and fixed by leaf springs (Ti-6Al-4V). Segregated panel area is limited up to 1 m2, and the design of the leaf spring is considered to reduce thermal stress, and to withstand the mechanical loads of plasma disruption and seismic loads. The coolant is 80K gas helium and both sides of panel are covered with multi-layers super insulation (SI) to reduce heat load (radiation) up to 1/100. Fraction of non-covered region is assumed to be 2% due to many port-joints and supports for the vacuum vessel. Total heat load for inner surface of cryostat (600m2) is 9kW and the heat load for the port-joints (-300m2) is assumed up to 9kW. The operational pressure of the cryostat is required to keep less than 1E-2 Pa and about 100,000 m2 of structural surfaces is considered for exhaust system specification.
Another role of the cryostat is the radiation protection. Biological shielding up to 10 micro-Sv/h (for maintenance acceptance) is required of the cryostat surface after the 10 years operation. Thus the cryostat consists of boron (2 wt%) doped concrete of 220 mm thickness and structural SS304 of total 40 mm thickness. The concrete reduces the air activation (41Ar) in the torus hall by 90% rather than the normal one by the thermal neutron absorption of boron.

THE MK III ACTIVELY COOLED DUCT LINER FOR THE JET NEUTRAL BEAM LINE: THERMO-MECHANICAL PERFORMANCES AND LIFETIME ESTIMATION

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This paper describes the analyses performed to investigate and validate the proposed design for the updated JET MKIII duct liner. The study was carried out in the frame of the NBE EP2 (Neutral Beam Enhancement Project 2) whose main objectives were to increase the deuterium neutral beam power, to increase the pulse duration to 20 s (from the actual 10 s) and to improve the beam reliability and availability.
The present inertial cooled duct protection would reach unacceptable temperatures if subjected to the integrated heat load envisaged for the new neutral beam parameters. An actively cooled duct protection has therefore been proposed. The design relies on hypervapotron cooling elements whose technology has been proven over many years and is used successfully on both JET and MAST. The proposed design results from the compromise between the allowable space and the requirement to keep the surface temperature lower than 200 °C, avoiding the gas re-emission and re-ionisation run away effect. Additionally the liner must fit in the existing main port assembly and be supported by the existing conical support and must have the same beam facing profile as the present one to avoid further encroachment. Each liner is constituted by six CuCrZr horizontal fingers 1.2 m long and shaped to match the duct beam profile. A ‘hairpin’ hypervapotron channel is machined in each finger, so the inlet/outlet collectors are located at the duct entrance. A 1.5 mm poloidal gap is foreseen between two adjacent fingers in order to allow for thermal expansion. The fingers sidewalls are profiled to prevent beam streaming through the gap and hitting the conical support structure. Each finger is connected to the conical support through three flexible attachments allowing a limited deformation under thermal loads and withstanding the torques acting on the finger in case of a disruption. In order to assess the thermal-hydraulic and thermo-mechanical duct liner performance, three thermal loading scenarios were defined and their probability of occurrence estimated on the basis of the previous JET experimental campaigns. The torques and acceleration due to disruptions were taken into account when considering the component clearances and dimensions. Two different scale finite element models have been assessed. Due to the aspect ratio of the hypervapotron fingers (~1.2 m) and a typical hypervapotron section, it is not practical to build a unique FE model to evaluate instantaneously the overall reactions, displacements and the concentrated stresses.
FE models representing an entire finger have therefore been used to estimate the overall displacements and the relative effect of the different loads (surface heat flux, internal pressure, disruption torques and accelerations) as well as the maximum stresses location and the displacements in the various loading scenario.
The sub-modelling technique has been applied to calculate the peak stresses and deformations on the most loaded sections appropriately meshed.
The consistency with the assumed dimensioning rules has been checked and the fatigue lifetime has been estimated.
This work was performed under the European Fusion Development Agreement, and is funded jointly by the UK EPSRC and EURATOM.

K-0002 (Invited)

DEMO CONCEPTS AND THEIR ROLES WITHIN THE FUSION PROGRAMME

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In the past years, the international fusion community has developed models of fusion power plants, which were extremely useful in showing the key advantages of fusion energy and pointing out he areas of development. The present view is that between ITER and such power plants (even of "first of kind" type), there is a need for one or two intermediate steps.
The need to have a "fast rack" towards such a fusion reactor, suggested that the steps after ITER, which are usually considered to be a Demonstration power plant followed by a Prototypical one, could be combines into one known as a DEMO. DEMO would then be a device capable of producing electricity, paving the way towards fusion power plants which would be economically viable.
This talk will outline the DEMO concepts as the necessary physics and technological extrapolation from the envisaged future steps (ITER, IFMIF) will be discussed. It will attempt to provide a coverage of the different concepts developed by various countries, The key issues, as foreseen today, and their implications for the programme will be highlighted.

P2-0001 (Invited)

EUROPEAN DEMO DESIGN AND MAINTENANCE STRATEGY

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The paper will outline the DEMO design activities in Europe, the rationale adopted for converging on the proposed design, and, in particular, the options being considered for the maintenance strategy.
The first phase of the design process began with a definition of parameters, when a number of key compromises were made between physics, technology and engineering factors. The divertor is a typical example of this process, where the physics suggests that the higher the heat load, the more compact the device will be; but technology defines the structural and functional materials available, and engineering considerations clarify the allowable heat load, depending on the cooling medium. Another example is the magnet system; here the number of coils has to be chosen based on geometry and ripple; and the decision between high or low temperature superconducting technology has to be made.
Following these early design choices, the key aspect considered in the design is the overall integration of the project, and a great deal of emphasis is placed on this activity.
Most of the design choices are based on analysis and/or supporting R&D, but one key issue which still remains to be resolved is the maintenance strategy. The paper will outline the maintenance requirements, summarise the various concepts considered to date, and indicate why none of these concepts fully satisfies the requirements.

P2-0002 (Invited)

THE PATH FROM ITER TO A POWER PLANT – INITIAL RESULTS FROM THE ARIES "PATHWAYS" PROGRAM

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The US national power plant studies program, ARIES, has initiated a 3-year integrated study, called the "Pathways Program" to investigate what the fusion program needs to do, in addition to successful operation of the ITER, in order to transform fusion into a commercial reality. The US power industry and regulatory agencies view the demonstration power plant, DEMO, as a device which is build and operated by industry, possibly with government participation, to demonstrate the commercial readiness of fusion power. As such, the "Pathways" programs will investigate what is needed, in addition to successful operation of ITER, to convince industry to move forward with a fusion DEMO.

While many reports exists that provide a strategic view of the needs for fusion development; in the ITER era, a much more detailed view is needed to provide the necessary information for program planning. By comparing the anticipated results from ITER and existing facilities with the requirements for a power plant in the first phase of the Pathways study, we will develop a comprehensive list of remaining R&D items for developing fusion, will identify metrics for distributing resources among R&D issues, and will identify which of those items can/should be done in existing or simulation facilities. In the second phase of the study, we will develop potential embodiments for the fusion test facility (ies) and explore their cost/performance parametrically. An important by-product of this study is the identification of key R&D issues that can be performed and resolved in existing facilities to make the fusion facility cheaper and/or a higher performance device.

This paper will summarize the results from the first phase of our study. We have adopted a "holistic" or integrated approach with the focus on the needs of the customer. In such an approach, the remaining R&D should generate all of the information needed by industry to move forward with the DEMO, i.e., data needed to convince power industry to invest in a fusion system, the licensing authority to license such a device, etc Through this approach we have identified many operational and licensing issues which had not previously received sufficient attention. An industrial advisory committee is guiding us through this process.

Given the limited resources available for fusion development, it is essential that we develop metrics for distributing resources among R&D issues. We have revisited various ARIES tokamak design and investigated the parameter space available for a fusion power plant and the impact of various constraints. Only by focusing on the final product, one can arrive at a satisfactory metric for prioritizing R&D (e.g., is it more prudent to push for a higher plasma or a high-efficiency blanket?). Of course, the resulting improvement in the attractiveness of the final product, should be judged against additional resources and risk associated with that particular R&D.

P2-0003 (Invited)

JAPANESE PERSPECTIVE OF FUSION NUCLEAR TECHNOLOGY FROM ITER TO DEMO

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The world fusion community is now launching construction of ITER, the first nuclear-grade fusion machine in the world. In parallel to the ITER program, Broader Approach (BA) activities are to be initiated in this year by EU and Japan, mainly at Rokkasho BA site in Japan, as complementary activities to ITER toward DEMO. The BA activities include IFMIF-EVEDA (International Fusion Materials Irradiation Facility-Engineering Validation and Engineering Design Activities) and DEMO design activities with generic technology R&Ds, both of which are critical to the rapid development of DEMO and commercial fusion power plants.

The Atomic Energy Commission of Japan reviewed on-going third phase fusion program and issued the results of the review, `On the policy of Nuclear Fusion Research and Development' in November 2005. In this report, it is anticipated that the ITER will be made operational in a decade and the programmatic objective can be met in the succeeding seven or eight years. Under this condition, the report presents a roadmap toward the DEMO and beyond and R&D items on fusion nuclear technology, indispensable for fusion energy utilization, are re-aligned.

In the present paper, Japanese view and policy on ITER and beyond will be summarized mainly from the viewpoints of nuclear fusion technology, and a minimum set of R&D elements on fusion nuclear technology, essential for fusion energy utilization, will be presented.

RUSSIAN DESIGN STUDIES OF THE DEMO-S DEMONSTRATION FUSION POWER REACTOR

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Different concepts for a fusion power plant have been studied in Russia since 1975. Researchers have considered power facilities using tokamaks, stellarators and inertial fusion devices. Tokamak reactors appear the most promising at this stage of science development. The studies have not revealed any essential advantages of stellarator-reactors over tokamak-reactors. The lack of physical database hinders the design teams` ability to engineer reactors based on mirror machines. Creation of an inertial fusion reactor proved to be much more difficult than building a tokamak reactor. Application of fusion reactors for generation of electricity, production of domestic and industrial heat, hydrogen production, transmutation of non-fissionable isotopes into fissionable ones, water desalination, and burning out of minor actinides was considered. Conceptual design studies of a tokamak-based demonstration fusion reactor have been carried out since 1991. The preferred concept was selected, which was a steady-state operating tokamak with superconducting magnets, one-null divertor configuration and a high contribution of bootstrap current into plasma current drive. The general reactor layout was determined. Plasma characteristics were optimized. Two most attractive blanket concepts were analyzed: (1) a He-cooled ceramic (Li4SiO4) design for tritium breeding, using ferritic steel as structural material, and (2) a blanket using liquid Li as tritium breeding material and coolant and a V-Cr-Ti alloy as structural material. The studies were supported by neutronic, heat-hydraulic and mechanical calculations. A conventional type of water or Li cooled divertor targets with maximum heat load of ~10 MW/m2 was chosen. Blankets of both types require Be as a neutron multiplier and have to be replaced after the integral fusion neutron load on the first wall reaches 10 MW/m2. Heat to electricity conversion schemes enable operation with net efficiency of 34% for the He-coolant design and 40% for the liquid Li one. Aspects of radioactive waste management and scarce materials refabrication are considered. In particular, a radiochemical extraction technology for separation of V alloy components and their purification from activation products after reactor decommissioning was developed and tested on activated specimens in laboratory (stationary) conditions. The technology enables hands-on recycling of alloy components. The results of the study did not allow to select the best blanket concept for further development. They revealed problems that should be solved at the next stages of the research: the attainment of the reactor availability of at least 0.6, validation of the arrangements aiming at the decrease of heat loads on divertor plates. In the case of the He-cooled blanket, the efforts should be focused on increasing both the tritium breeding ratio and plant efficiency; in the case of the liquid Li design, they should concentrate on the validation of the design values of pressure loss and on the enhancement of the efficiency of the neutron shielding.

DEMO DEVELOPMENT STRATEGY BASED ON CHINA FPP PROGRAM

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The DEMO in China is to demonstrate the safety, reliability and environment feasibility of the fusion power plants, while to demonstrate the prospective economic feasibility of the commercial fusion power plants. Considering that there is still a long way to go towards an economically competitive commercial power plant, DEMO in China should be an indispensable step prior to the commercial one. Two options of breeding blanket with ceramic and lead lithium breeders might be chosen as DEMO concepts under the conditions of meeting the requirement of the neutronics, thermal-hydraulics and mechanics aspects. The DEMO development strategy, related R&D activies, based on China fusion power plant (FPP) program are presented.

S3-0003 (Invited)

A STRATEGIC PLAN OF KOREA FOR DEVELOPING FUSION ENERGY BEYOND ITER

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Since Korea joined ITER as they were developing a fully super-conducting tokamak, KSTAR, the national agenda for developing fusion energy was renewed and focused on the clear mission - developing a self-reliance program for a fusion demonstration device. A new technical roadmap was developed toward self-reliance of technologies for constructing a fusion power plant in the similar time window as in EU, Japan and US. This looks tough and formidable but Korea would take advantage of their well-established nuclear technologies in power plant design, construction and operation. The Government and the Legislature recognized importance of this effort by establishing a special law for supporting it practically and for giving it a high priority in their R&D agenda. Under this favorable environment, various scenarios for finding the most effective and efficient path to achieve the goal were reviewed.
Some of the features of the results of this review will be presented.

S3-0004 (Invited)

STRATEGY FOR THE INDIAN DEMO DESIGN

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The Indian scenario for the development of DEMO will be presented. The advantage of joining ITER and its consequences on the pace of fusion technology development in the country will be discussed. The strategy for the DEMO development is to build prototypes for proving the enabling technologies and to perform their integrated tests in an intermediate size tokamak (SST-2 with R0 ~4.4 m, Q~4) with modest fusion power. This will allow an experience with D-T cycle apart from high-power long-pulse operation. A crucial step will be offered by SST-2 for the human resource development for future. It will also evolve remote handling technology and several test facilities. Development of suitable structural materials and fusion-blankets will be major activities in parallel, which are receiving a boost due to India`s participation in the Test Blanket Module activity. The DEMO machine will have R0 ~7 m and Q ~20 approximately and will share the facilities setup for SST-2. At present, design modeling activities are underway for both the machines. From a programmatic point of view, a networked research activity is being kick-started by initiating a National Fusion Programme.

S3-0005

DEVELOPMENT OF TOKAMAK REACTOR SYSTEM CODE AND CONCEPTUAL STUDIES OF DEMO WITH HE COOLED MOLTEN LI BLANKET

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To develop the concepts of fusion power plants and identify the design parameters, we have been developing the tokamak reactor system code. The system code can take into account a wide range of plasma physics and technology effects simultaneously and it can be used to find design parameters which optimize the given figure of merits. The outcome of the system studies using the system codeis to identify which areas of plasma physics and technologies and to what extent should be developed for realization of a given fusion power plant concepts.
As an application of the tokamak reactor system code, we investigate the performance of DEMO for early realization with a limited extension from the plasma physics and technology used in the design of the ITER. Main requirements for DEMO are selected as: 1) to demonstrate tritium self-sufficiency, 2) to generate net electricity, and 3) for steady-state operation. The size of plasma is assumed to be same as that of ITER and the plasma parameters which characterize the performance, i.e. normalized beta value, beta_N, confinement improvement factor for the H-mode, H and the ratio of the Greenwald density limit n/n_G are assumed to be improved beyond those of ITER: beta_N > 2.0, H > 1.0 and n/n_G > 1.0. Tritium self-sufficiency is provided by the He Cooled Molten Lithium (HCML) blanket with the total thickness of 2.5 m including the shield.
With n/n_G > 1.2, net electric power bigger than 500 MW is possible with beta_N > 4.0 and H > 1.2. To access operation space for higher electric power, main restrictions are given by the divertor heat load and the steady-state operation requirements. Developments in both plasma physics and technology are required to handle high heat loadand to increase the current drive efficiency.

S3-0006 (Invited)

DIVERTOR CONCEPTUAL DESIGNS FOR A FUSION POWER PLANT

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The development of a divertor concept for post-ITER fusion power plants is deemed to be an urgent task to meet the EU Fast Track scenario. Developing a divertor is particularly challenging due to the wide range of requirements to be met including the high incident peak heat flux, the blanket design with which the divertor has to be integrated, sputtering erosion of the plasma-facing material caused by the incident a particles, radiation effects on the properties of structural materials, and efficient recovery and conversion of the divertor thermal power (~15% of the total fusion thermal power) by maximizing the coolant operating temperature while minimizing the pumping power.

In the course of the EU PPCS, three near-term (A, B and AB) and two advanced power plant models (C, D) were investigated. Model A utilizes a water-cooled lead-lithium (WCLL) blanket and a water-cooled divertor with a peak heat flux of 15 MW/m2. Model B uses a He-cooled ceramics/beryllium pebble bed (HCPB) blanket and a He-cooled divertor concept (10 MW/m2). Model AB uses a He-cooled lithium-lead (HCLL) blanket and a He-cooled divertor concept (10 MW/m2). Model C is based on a dual-coolant (DC) blanket (lead/lithium self-cooled bulk and He-cooled structures) and a He-cooled divertor (10 MW/m2). Model D employs a self-cooled lead/lithium (SCLL) blanket and lead-lithium-cooled divertor (5 MW/m2). The values in parenthesis correspond to the maximum peak heat fluxes required.

It can be noted that the helium-cooled divertor is used in most of the EU plant models; it has also been proposed for the US ARIES-CS reactor study. Since 2002, it has been investigated extensively in Europe under the PPCS with the goal of reaching a maximum heat flux of at least 10 MW/m2. Work has covered many areas including conceptual design, analysis, material and fabrication issues, and experiments. Generally, the helium-cooled divertor is considered to be a suitable solution for fusion power plants, as it avoids the use of water cooling associated with He-cooled Be-ceramic blanket systems that would lead to considerable safety concerns (e.g. steam-beryllium reaction and H production). Moreover, it allows for a relatively high gas outlet temperature and, hence, a high thermal efficiency of the power conversion systems.

This paper provides an overview of the development of different conceptual designs of divertors for fusion power plants; their advantages and disadvantages and expected performance are outlined and discussed. Emphasis is placed on summarizing the status and progress of R&D associated with He-cooled divertor design in Europe and USA.

THE WAY FROM ITER TO THE WALL MATERIAL SELECTION FOR DEMO

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The present fusion experiences are largely based on devices with graphite as first wall material. The graphite choice allows for a large operational space in all plasma scenarios, with small detrimental effect of impurities on the main plasma performance (low Z) and for power loads in transients and steady state exceeding occasionally the material limits. However, the chemical affinity of graphite with hydrogen results in large carbon erosion, migration and redeposition, which would lead to unacceptable T retention in ITER, based on extrapolation from present devices. The ITER wall material choice is determined by aim to combine the advantages of C (no melting) at the high heat flux areas, of Be on the large scale first wall (low Z) and of W on the upper divertor baffle (low erosion, large lifetime) while minimising simultaneously the critical issues of C (T retention) and of W (plasma contamination). This material choice will be tested first in the new JET ITER like wall project which serves then to define the ITER operational conditions such that they are compatible with the wall requirements and scientific goals of ITER. The wall choice in ITER is still largely determined by the possibility for detrimental conditions with respect to transient power loads, the main plasma contamination with high Z impurities and also the relatively low duty cycle and moderate amount of neutrons. In contrast, for DEMO the development of plasma control must allow the use of first wall materials that are optimised fully by the materials requirements of mechanical stability , low T retention, low erosion (large lifetime) and low activation. This is the challenge in fusion both for plasma physics and control and material physics.

TESTING OF PLASMA FACING MATERIALS AND COMPONENTS AT HFR PETTEN

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The operation of ITER will depend largely on the reliable performance of its severely loaded in-vessel components. With ITER the fusion community
moves definitely into the era of fusion nuclear technology. This requires the qualification of key components, through materials irradiations, out-of-pile tests, in-pile tests and rigorous analyses.
For the qualification of plasma facing components an irradiation facility is being developed to simulate simultaneously the effect of thermal fatigue and cyclic neutron irradiation loads. Three first wall mock-ups with beryllium armor tiles will be irradiated to about 1 dpa in beryllium with parallel thermal fatigue testing for 30,000 cycles. The temperatures, stress distributions and stress amplitudes at the Be/CuCrZr interface of the mock-ups will be representative for the ITER First Wall panels.
For this objective the FW mocks-ups subjected to thermal fatigue will be
integrated with high density (W) plates on the Be-side to provide the heat flux by nuclear heating. The assembly will be placed in the pool-side facility of the HFR and thermal cycling is then arranged by mechanical movement towards and from the core box.
To verify the thermo-mechanical model and the nuclear analysis a pilot rig is being designed. The pilot rig will contain a single well instrumented mock-up and replaceable neutron flux monitor sets. Explosion bonding of tungsten is a potential manufacturing technique for areas with low to medium heat fluxes, like the ITER dome and baffle area.
The aim of this study is to demonstrate that these large area components can be successfully coated by this technique. In the exploratory stage W foil and/or plate were joined by explosion welding to substrates of SS316L and CuCrZr and to itself. In the current stage small scale welding trials are examined with NDT techniques as US and Eddy Current. In the final stage medium scale test samples will be produced. They are foreseen to be examined with High Heat Flux testing.
The development of advanced plasma facing materials focuses on heat sink and radiation resistant materials. The outline of work performed in the ExtreMat Integrated Project will be shown. This comprises a.o. nano-structured W, Chromium-Rhenium alloys, brazed joints and coated systems for primary wall and divertor applications.

S3-0009 (Invited)

PLANS FOR IGNITION EXPERIMENTS ON THE NATIONAL IGNITION FACILITY*

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The National Ignition Facility (NIF) is a 192 beam Nd-glass laser facility presently under construction at LLNL for performing ignition experiments for inertial confinement fusion (ICF) and experiments studying high energy density (HED) science. NIF will produce 1.8 MJ, 500 TW of ultraviolet light making it the world's largest and most powerful laser system. NIF will be the world's preeminent facility for the study of matter at extreme temperatures and densities producing and for developing ICF. The ignition studies will be the next important step in developing inertial fusion energy.

The NIF Project is over 90% complete and scheduled for completion in 2009. The building and nearly the entire beam path have been completed. Fig. 1 shows the beam path in one of the two laser bays. The Project is presently installing the optics and electronics and commissioning the beams. Over half of the optical and electronics components needed to complete the Project have been installed. One cluster of 48 beams has been commissioned in the laser bay with the demonstrated capability of producing 1000 kJ of 1053 nm light (1_), nearly ten times the capability of Nova or Omega, the previous largest laser systems. In addition, experiments using one beam have demonstrated that NIF can meet all of its performance goals.

A detailed plan called the National Ignition Campaign (NIC) has been developed to begin ignition experiments in 2010. The plan includes the target physics and the equipment such as diagnostics, cryogenic target manipulator and user optics required for the ignition experiment. Target designs have been developed that calculate to ignite at energy as low as 1 MJ. Experiments using the OMEGA laser at the University of Rochester are validating these designs. Development of manufacturing capability is well under way for producing these targets to the required tolerances. Diagnostics and other support equipment is being designed and fabricated to perform the ignition experiments.

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3-D NUCLEAR ANALYSIS OF THE FINAL OPTICS OF A LASER DRIVEN FUSION POWER PLANT

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In the High Average Power Laser (HAPL) program, power plant designs are assessed with 350 MJ yield targets driven by 40 KrF laser beams. The final optics system that focuses the laser onto the target includes a grazing incidence metallic mirror (GIMM) located at 24 m from the target with 85¡ angle of incidence. The GIMM is in direct line of sight of the target and has a 50 microns thick aluminum coating. Several options were considered for the substrate material. We performed three-dimensional (3-D) neutronics calculations to assess the impact of the GIMM design options on the nuclear environment at the dielectric focusing and turning mirrors. We used the recently developed MCNPX-CGM Monte Carlo code that allows performing the neutronics calculations directly in the exact CAD model. The most recent continuous energy fusion evaluated nuclear data library (FENDL-2.1) was used. One of the 40 beamlines was modeled with surrounding reflective boundaries. We considered beam duct configuration modifications such as utilizing neutron traps behind the mirrors to reduce radiation streaming. Several variance reduction techniques were utilized to reduce the statistical uncertainties. The results indicate that material choice and thickness for the GIMM impact the nuclear environment at all mirrors. The neutron flux and nuclear heating at the dielectric mirrors are a factor of ~1.6 higher when AlBeMet is used instead of SiC as substrate in the GIMM. The fast neutron flux decreases by about two orders of magnitude as one moves from the GIMM to the focusing mirror with an additional two orders of magnitude attenuation at the turning mirror accompanied with significant spectrum softening. In this paper, the details of the analysis and results will be presented and the expected optics lifetime will be assessed.

S4-0001 (Invited)

HIGH TEMPERATURE DEMO BLANKET CONCEPT FOR HYDROGEN PRODUCTION

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Hydrogen is considered as the most potential energy carrier in the near future and can be produced from fusion nuclear energy by several means. Water electrolysis can be served by fusion electrical energy, the steam reforming reaction and the thermochemical water-splitting process can be served by fusion thermal energy. The works about steam reforming with biomass waste and the thermochemical water-splitting with S-I cycle have been extensively investigated in the world. The radiation chemical process and plasma chemical process for hydrogen production also have been reported recently.

The fusion energy can be improved by the development of high performance reactor. Some high temperature reactors based on SiCf/SiC composites have been developed, such as ARIES-AT, with high outlet temperature about 1100 oC suitable for hydrogen thermal processes. But several issues for the SiCf/SiC composites including the uncertainty about behavior and performance at high temperature and under irradiation, the fabrication and joining technology have been addressed to limit the current development and application of high temperature blanket.

RAFM steel remains presently the most promising structural material for breeding blanket with great technological maturity. An innovative high temperature liquid blanket concept is proposed based on RAFM steel as structural material and LiPb as tritium breeder. A special design is designated to obtain the high temperature LiPb about 1000 oC far more higher than the RAFM temperature limit 550 oC, that is, the multi-layer flow channel inserts (MFCIs) made of the refractory material are placed inside the LiPb flow channels. Low temperature LiPb flows into the channel, meanders through the MFCIs. The temperature of the LiPb is increased step by step, at last it is exported from the blanket at the high outlet temperature.

The technology bases on the hydrogen production processes and the high temperature DEMO blanket development are reviewed and assessed in this paper at first. Then the conceptual design, performance analysis covering neutronics and thermal¨Chydraulics, safety and environmental impact, etc. of the novel high temperature blanket, and the assessment of hydrogen efficiency based on S-I thermochemical cycle are given. R&D needs are specified in the end, especially to avoid the neutron activation contamination and tritium contamination for hydrogen.

S4-0002 (Invited)

REVIEW OF BLANKET DESIGNS FOR ADVANCED FUSION REACTORS

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The dominating fraction of the power generated by fusion in the reactor is captured by neutron moderation in the blanket surrounding the plasma. From this, the efficiency of the fusion plant is predominated by the technologies applied to make electricity or hydrogen from the neutrons. Next to the blanket technology itself, also the compatibility with advanced power conversion systems and the coolant cycles have to be considered in detail. Furthermore, the different blanket concepts have to be compared in the fields of (i) overall system and development costs and (ii) risks including all related subsystems outside the reactor like tritium extraction, heat exchangers and power conversion, (iii) technical feasibility, (iv) reliability, (v) ease of manufacture, (vi) maintainability, (vii) compatibility with advanced reactor layouts (integration, neutron wall and surface heat loads) and (viii) safety. As fusion reactor power stations will have to compete with other types of central power stations, a too conservative approach will likely be not attractive enough in terms of cost of electricity to boost fusion technology. On the other side a too risky approach relying on tremendous budgets to solve severe technical issues might guide into a long unsuccessful scientific route.

The blanket concepts mainly addressed in this paper are advanced ceramic breeder concepts, dual coolant blankets as well as self cooled liquid metal and FliBe blankets. The important questions that will be addressed by the current paper are (i) can we draw a bottom line under the conceptual design and system descriptions of different concepts reviewed in this paper and conclude on the most promising concept(s) and (ii) what are the common issues to be solved independently from individual design and layout proposals to define a feasible route towards advanced fusion reactors.

S4-0003 (Invited)

RECENT RESEARCH AND DEVELOPMENT FOR THE US DUAL-COOLANT LEAD-LITHIUM BLANKET

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The Dual-Coolant Lead-Lithium, or DCLL, blanket concept with flow channel inserts (FCIs) serving as thermal and electrical insulator was initiated in the ARIES-ST study and became the reference blanket in ARIES-CS. It has been also investigated under the APEX program and Test Blanket Module effort in the US, and in the EU as well. The DCLL is constructed from Reduced Activation Ferritic/Martensitic (RAF/M) steel with the first wall and all internal structures cooled by high pressure helium to a temperature below the allowable operating temperatures for RAF/M steel. The flowing lead-lithium (PbLi) coolant/breeder, however, is thermally and electrically isolated from the He-cooled structures by loose-fitting, flow channel inserts (FCIs) that serve no structural function, but which are compatible with PbLi at much high temperatures that the RAF/M steel. With such an arrangement, the DCLL He coolant emerges from the blanket at typical temperatures ~450C, but with the PbLi coolant emerging with outlet temperatures ~700C for improved thermal efficiency. The circulation speed of the PbLi is still fast enough that tritium partial pressure can be controlled to a relatively low level with a high efficiency extraction technique.

For the DCLL blanket itself, the flow channel insert is the critical piece. FCIs must have low electrical and thermal conductivity and be compatible with PbLi at elevated temperatures. FCIs must retain structural integrity and desirable properties even under irradiation and large temperature gradients during operation. FCIs must not fail in such a way that PbLi enters the FCI and changes its electrical conductivity appreciably. Another important issue for the DCLL is the development of a suitable tritium extraction from PbLi to achieve a tritium partial pressure < 1Pa, facilitating decisive tritium control.

In this paper, the state of DCLL development in the US is presented including recent design modifications and results from recent R&D efforts. Such R&D includes the progress on development of SiC/SiC composites and SiC foams as FCI candidates; electrical and thermal conductivities for FCI materials, PbLi capability and infiltration studies, simulations MHD flow characteristics and of resultant temperature distributions; and the analysis of FCI stress states and deformation based on these thermal loads. In addition, tritium extraction from PbLi based on a vacuum permeator is shown to have the potential to achieve desired tritium control. A discussion of unresolved DCLL issues and future R&D needs and plans in the US is also presented.

S4-0008

PRELIMINARY DESIGN OF INDIAN TEST BLANKET MODULE FOR ITER

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Indian Test Blanket Module (TBM) program in ITER is one of the major steps in its fusion reactor program towards DEMO and future Fusion Power Reactor (FPR) vision. Along with the DEMO design, two blanket concepts were chosen for the development and testing in ITER. The primary blanket concept is based on liquid breeder type known as Lead-Lithium cooled Ceramic Breeder (LLCB) TBM which was proposed recently and a helium-cooled solid breeder concept with ferritic steel structure and Be neutron multiplier, but as a sub-module type.

Presently the prime focus is on the design and analysis of the LLCB TBM, which is based on the Pb-Li eutectic flow cooling the ceramic breeder encased in partition of Ferritic steels structure cooled by high pressure helium. The ceramic breeder beds, which are filled with lithium titanate pebbles, act as partition for coolant and essentially become a part of structure. The attractive feature of the TBM is Pb-Li acts as the coolant and breeder material in addition to the ceramic breeder material. The ceramic pebble zones will be purged by a low pressure Helium flow for tritium extraction. The tritium produced in Pb-Li will be extracted by separate external ancillary system. The R&D activities are being initiated in all critical areas related to DEMO relevant blanket concepts in order to test the TBM in ITER.

In this paper, the design description, the performance analysis (tritium breeding ratio, thermal behaviour under DEMO conditions, etc.) and the related ancillary systems for LLCB TBM will be presented.

S4-0009

TECHNICAL ISSUES OF RAFMS FOR THE FABRICATION OF ITER TEST BLANKET MODULE

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Reduced activation ferritic/martensitic steels (RAFMs) are recognized as the primary candidate structural materials for fusion blanket systems, as it has they have been developed based on massive industrial experience of ferritic/martensitic steel replacing Mo and Nb of high chromium heat resistant martensitic steels (such as modified 9Cr-1Mo) with W and Ta, respectively. F82H and JLF-1 are RAFMs, which have been developed and studied in Japan and the various effects of irradiation were reported. F82H is designed with emphasis on high temperature property and weldablility, and was provided and evaluated in various countries as a part of the IEA fusion materials development collaboration. The JAEA/US collaboration program also has been conducted with the emphasis on irradiation effects of F82H. Now, among the existing database for RAFMs the most extensive one is that for F82H. The objective of this paper is to review the R&D status of F82H and to identify the key technical issues for the fabrication of ITER Test Blanket Module (TBM) suggested from the recent achievements in Japan.

It is desirable to make the status of RAFMs equivalent to commercial steels to use RAFMs as the ITER-TBM structural material. This would require demonstrating the reproducibility and weldability as well as providing the database. The excellent reproducibility of F82H has been demonstrated with four 5-ton-heats, and two of them were provided as F82H-IEA heats. It has been also proved that F82H could be provided as plates (thickness of 1.5 to 55mm), pipes and rectangular tubes. It is also important to have the excellent weldability as the TBM has about 300m length of weld line, and it was proved through TIG, EB and YAG weld test performed in air atmosphere. Various mechanical and microstructural data have been accumulated including long-term tests such as creep rupture tests and aging tests.

Although F82H is a well-perceived RAFM as the ITER-TBM structural material, some issues are remaining to be examined to assure its applicability even under the most severe operation scenario. ITER will be operated in the pulsed mode with cycles up to 30,000, and severe plasma disruptions are expected to occur 100 times per year, suggesting that the structural materials will experience fatigue and/or creep fatigue loading. The fatigue softening would be an issue to degrade the strength of RAFMs during the ITER operation. Another issue would be the effects of Ta. Ta was added to all RAFMs to improve fracture toughness and creep strength, but recent work found it tends to form inclusions and causes toughness inhomogeneity, and increases the possibility of hot cracking during welding. To overcome these issues, it is essential to obtain a good understanding of the detailed mechanical properties and design TBM carefully based on that.

SYNERGIES IN THE DESIGN AND DEVELOPMENT OF FUSION AND GENERATION IV FISSION REACTORS

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Future fusion reactors or systems and Generation IV fission reactors are designed and developed in worldwide programmes mostly involving the same partners to investigate and assess their potential for realisation and contribution to meet the future energy needs beyond 2030. Huge scientific and financial effort is necessary to meet these objectives. First programmes have been launched in Generation IV International Forum (GIF) for fission and in the Broader Approach for fusion reactor system development. Except the physics basis for the energy source, future fusion and fission reactors, in particular those with fast neutron core face similar design issues and development needs. Therefore the call for the identification of synergies became evident. Beyond ITER cooled by water, future fusion reactors or systems will be designed for helium and liquid metal cooling and higher temperatures similar to those proposed for some of the six fission reactor concepts in GIF with their diverse coolants. Beside materials developments which are not discussed in this paper, design and performance of components and systems related to the diverse coolants including lifetime and maintenance aspects might offer significant potentials for synergies. Furthermore, the use of process heat for applications in addition to electricity production as well as their safety approaches might create synergistic design and development programmes. Therefore an early identification of possible synergies in the relevant programmes should be endorsed to minimise the effort for future power plants in terms of investments and resources.

In addition to a general overview of a possible synergistic work programme which promotes the interaction between fusion and fission programmes towards an integrated organisation of their design and R&D programmes, some specific remarks will be given for joint design tools, numerical code systems and joint experiments in support of common technologies.

S4-0006 (Invited)

LIQUID METAL COOLING ISSUES FOR FUSION AND FISSION

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Liquid metal application for nucear power plants was initiated in the design of fast reactors using sodium or lead bismuth, and are very famous for sodium fast breeder reactor and lead bismuth fast reactor. In a course of development of FBR, Na and NaK were carefully compared and the former was chosen as the candidate coolant, followed by various Na cooled test reactors. PbBi fast reactor was put into practical use as a engine for nuclear ship, following research activities as a thermal reactor with PbBi coolant. In the nuclear fusion application, liquid metals of Li or LiPb will be used as a coolant and/or tritium breeder in comparison with water and helium gas.
It is well known that in a nuclear reactor, materials are required to function as moderator and coolant. Though these fucntions are satisfied with separate materials, it is very reasonable to combine these functions to one materials, such as water or sodium, as is rearized by LWR or Na-FBR The economical success of these reactors depend on the selection of coolant material that works as neutron moderator at the same time. This design option resulted in a great reduction of the size of system and in an increase of the system efficiency.
On the analogy of this history, fusion coolant might be cosolidated in future to the material that works as the coolant and T breeder, such as Li or LiPb, and not in water or helium which need separate breeding material in addition to coolant. However, the MHD flow problem does not exist in nuclear power plants and the prospect for the solution of this problem will largely affect conclusions.
The other example of liquid metal application, is the liquid metal target, represented by Li target for IFMIF and PbBi target for ADS. The technology required in IFMIF may be quite different, and may rather be nearer to those for inertia fusion power plant.

S4-0007 (Invited)

FUSION-FISSION HYBRIDS FOR NUCLEAR WASTE TRANSMUTATION: A SYNERGISTIC STEP BETWEEN GEN-IV AND FUSION REACTORS

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Due to increasing concerns over the buildup of long-lived transuranic isotopes in spent nuclear fuel waste, attention has been given in recent years to technologies that can burn up these species. Fast reactors are currently the technology of choice for this goal, but it will take fleets of fast reactors to burn transuranics as fast as light water reactors produce them. Since fast reactors are expected to cost more than light water reactors, it is unlikely that fleets will be built any time soon. Fusion transmutation of waste offers an alternative to fast reactors offers some key advantages in the fuel cycle.

Sandia National Laboratories is investigating the use of a Z-Pinch fusion driver for transmuting waste. Relatively modest fusion requirements on the order of 20 MW can be used to drive a sub-critical transuranic blanket that produces 3000 MWth. The actinides are is a fluid form which eliminates the need for expensive fuel fabrication and allows for continuous refueling, removal of fission products, and tritium breeding to sustain the fusion driver. This reactor has the capability of burning 1280 kg of actinides per year while at the same time producing a significant amount of power.

There are two key advantages of burning actinides in a fluid, sub-critical blanket. The first is that there is no need to have fertile fuel (like 238U) in the blanket. This means that the blanket only contains transuranics, so it burns waste with the maximum efficiency. The second advantage is that the blanket can be fueled with virtually any transuranic mix depending on the fuel cycle of the future. Therefore it has much more flexibility than a fast reactor and can, for example, be designed to burn only Np/Am/Cm.

A more realistic and logical approach to waste reduction in the fuel cycle is to burn Pu in existing light water reactors, while building one fusion transmutation reactor to only burn the minor actinides Np/Am/Cm. After a few decades when the limit of Pu recycle in light water reactors is reached, additional transmuters can be built to take care of the spent Pu fuel. This strategy ultimately requires much fewer fast transmutation systems to be built while at the same time achieving the same waste reduction goals. Only a fusion-driven sub-critical transmutation reactor has the flexibility to achieve this fuel cycle since it is not possible to control a fast reactor that only uses Np/Am/Cm fuel. This application provides fusion with a useful application and valuable experience in the design of fusion reactor systems, so that one day we can achieve the ultimate in waste reduction: pure fusion energy.

S4-0004 (Invited)

GOALS, CHALLENGES, AND SUCCESSES OF MANAGING FUSION ACTIVATED MATERIALS

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After decades of designing magnetic and inertial fusion power plants, it is timely to develop a new framework for managing the activated materials generated during plant operation and after decommissioning - a framework that takes into account the lessons learned from numerous international fusion and fission studies and the environmental, political, and present reality in the U.S., EU, and Japan.

Since the inception of the fusion projects in the early 1970s, the majority of power plant designs have focused on the disposal of active materials in geological repositories as the main option for handling the replaceable and life-of-plant components, adopting the preferred fission waste management approach. It is becoming evident that future regulations for geological burial will be upgraded to assure tighter environmental controls. Along with the political difficulty of constructing new repositories worldwide, the current reality suggests reshaping all aspects of handling the continual stream of fusion active materials. There is a growing international effort in support of this new trend. Beginning in the mid 1990s and continuing to the present, fusion designs developed in Europe, U.S., and Japan have examined replacing the disposal option with more environmentally attractive approaches, redirecting their attention to recycling and clearance while continuing the development of materials with low activation potential. These options became more technically feasible in recent years with the development of radiation-hardened remote handling (RH) tools and the introduction of the clearance category for slightly radioactive materials by national and international nuclear agencies.

We applied all scenarios to selected fusion studies. While recycling and clearance appeared technically attractive and judged, in some cases, a must requirement to control the radwaste stream, the disposal scheme emerged as the preferred option for specific components for several reasons, including economics, occupational dose minimization, and chemical toxicity. This suggests that the technical and economic aspects, along with the environmental and safety related concerns, must all be addressed during the selection process of the most suitable waste management approach.

To enhance prospects for a successful management scheme, additional tasks received considerable attention during this collaborative study and will be highlighted in this paper. These include the key issues and challenges for disposal, recycling, and clearance, the development of very low impurities content materials, the limited capacity of existing repositories, the status of the recycling infrastructure, the development of advanced RH equipment, the notable discrepancies between the various clearance standards, the need for new guidelines for fusion-specific radioisotopes, the availability of a commercial market for cleared materials, and the acceptability of the nuclear industry to recyclable materials.

ANALYSIS OF SUBCRITICAL SYSTEM CORRESPONDING TO ENERGY AMPLIFIER

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Spallation neutron consist basically of an accelerator providing a beam of high-energy (> 100 Mev) protons o possibly heavier ions and suitable target of heavy-element material. Neutron yields and spectra were measured for protons and deuterons of intermediate energies on thick target at several laboratories.
In this work we have the purpose to do the analysis of subcritical system corresponding to Energy Amplifier. We know the accelerator current intensity and we can obtain the current intensity I in protons/s that irradiate the subcritical assembly. The source intensity corresponding to produced neutrons by proton is the product Sn=S(E) I, being n refered to neutrons, and S(E) is the ratio of produced neutrons by proton in 208Pb82 that is given by a numerical approximation of a graphical function due to Stankowsky et alii. We study the system of a subcritical reactor with cylindrical symmetry and finite dimensions. The behaviour of neutronic flux is determined by means of the time-dependent diffusion equation in which the source density is replaced by slowing-down density in the equation of age theory (doing the age = 0) and applying the adequate boundary and initial conditions. The neutronic spectrum in the spallation has been recently obtained. We indicate the results obtained. The neutronic flux is calculated, being r and z the radial and axial coordinates, t the time and E the proton energy. New symbols appear,the effective multiplication factor, the mean neutronic diffusion time and the bukling , corresponding to mode (m,n). We suppose that the infinite multiplication factor is (K)inf=0.98. For an energy interval of [200,1000] Mev, we have obtained the normalized neutronic flux in a fixed point , in function of time and proton energy. The volumetric fraction of fuel is obtained as a function of two variables that depend of nuclear parameters. The obtained normalized neutronic flux for a fixed proton energy , in function of r and z , is a solution , that correspond to a subcritical system, having the same shape that normalized neutronic flux corresponding to a critical system but a smaller value. The neutronic flux for the subcritical assembly have been calculated. We choose the point (r=0,z=0), as reference for the neutronic flux calculations. In this picture we can see the variation of neutronic flux in function of the time and the energy for intervals [0,1] seconds for time and [200,1000] Mev for energy. The flux rising in a lineal form in function of energy since values of time very low. All the efective multiplication values corresponding to different modes are lower than the unity ,showing the character of subcritical assembly.

MULTI-MODULES HCLL BREEDING BLANKET DESIGN FOR DEMO

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A new DEMO design activity has been started in EU since January 2005 after the completion of the EU Power Plant Conceptual Study (PPCS). Following the establishment of a DEMO physics basis and a number of scoping studies, the design activity associated with the main in-vessel components has been launched in 2006.
The assumed set of DEMO parameters include an electrical power output of 1000 MWe and a major radius of 7.5 m which have been selected with the aim to minimize the plant capital cost rather than the cost of electricity.
The assumed reference breeding blanket is the Helium-Cooled Lithium-Lead (HCLL) blanket. The design of the blanket internal components for the reference HCLL blanket module is derived from the PPCS/Model AB studies and from previous EU DEMO studies. It consists of a He-cooled Eurofer-steel box reinforced with toroidal and poloidal He-cooled Eurofer stiffeners. He-pressure is 8 MPa and inlet/outlet temperatures are 300°C/500°C. The Pb-16Li slowly flows within the grid in radial direction and is cooled by He flowing in radial-toroidal steel cooling plates immersed in the Pb-16Li. In the rear, the module box is closed by several steel plates acting as distributing/collecting chambers fro the He-coolant and the Pb-16Li.
Besides the description of the blanket design and performances corresponding to the new DEMO specifications, this paper focuses on the integration aspect of HCLL blanket using vertical Multi-Module Segments (MMS) maintenance scheme. 16 toroidal coils defining a vacuum vessel with 16 sectors and the maintenance will be performed vertically through the 16 upper ports. Each sector is formed by 2 inboard and 3 outboard banana-like MMS. Therefore, each upper port will allow the extraction of 5 MMS.
Each MMS has 6 modules which have in common the last two He-collecting chambers whose walls act as a strong back-plate structure for each banana-like MMS. The typical size of these modules is about 1x2m². Starting from specific shielding analyses, the radial built has been optimised with the aim to minimize the major radius. An original flexible attachment has been designed between segments and vacuum vessel, allowing free thermal expansion of segments and resisting to electromagnetic loads.
The principle, for each of the 16 sectors, is to hang up a set of 5 segments (3 on outboard and 2 in inboard), rigidified by specific devices in order to form an arch, to 4 vertical flexible bars (2 on outboard and 2 on inboard). Shear keys are allocated on vacuum vessel in order to limit electromagnetic loads consequences on the in vessel structures and the shield is permanently fixed to the vacuum vessel.
Moreover, the layout of He and PbLi feeding pipes and the upper port design have been studied taking into account maintenance scenario.

PS3-1003

LIFETIME PERFORMANCE OF HCPB POWER PLANT IN-VESSEL COMPONENTS USING HERCULES

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The PPCS plant models explore a range of physics and technology assumptions spanning from near-term, ITER-like assumptions to very advanced scenarios. They were named A to D following an order of increasing extrapolation from current expertise, and differ substantially in plasma parameters, electrical output, blanket and divertor technology. Plant model B (PMB) employs a helium-cooled, pebble bed (HCPB) blanket concept. The structural material is the reduced activation EUROFER grade steel. Creep, both thermal and irradiation, is the primary limiting mechanism for in-vessel component lifetime, with the bulk of cumulative damage due to power–on period. Lifetime improvement is possible if the time-to-failure at the full power–on stress level can be extended, which can be achieved through geometrical shape optimization. The helium-cooled concept is prone to fatigue limited lifetime, unless the plasma control is such that it limits the overpower transients and at the same time keeps their number to a minimum. Also, at higher temperatures, the structure becomes more susceptible to fatigue and even a small number of large duration events is capable of reducing lifetime significantly. Redesign of the in-vessel build could rely on a slimmer tritium generating blanket (~40cm) instead of ~50cm, with first wall acting as a "shielding" blanket of ~10cm thick. The concept relies on extending the first wall and breeding blanket lifetime significantly, by limiting creep and fatigue, without losing the breeding capability of TBR>1. HERCULES was used for a neutronics analysis and the extraction of engineering parameters. A comparison with previous creep-fatigue analyses of helium cooled blankets and first walls, show that this new concept is capable of doubling the lifetime, with adequate TBR~1.16 (original TBR~1.36). The concept needs further refinement and optimization, but it is promising to deliver not only engineering parameters but also economic performance.

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PS3-1004

DESIGN STUDY OF JT-60SA DIVERTOR FOR HIGH HEAT AND PARTICLE CONTROLLABILITY

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In steady-state high performance plasma over 41 MW/100 s in the JT-60SA tokamak [1], the heat and particle flux density on the divertor targets are considerably higher than those of existing devices such as JT-60U. A divertor modeling code, SOLDOR/NEUT2D [2], has been applied in order to optimiz the JT-60SA divertor design in such conditions. The heat load qheat on divertor target is estimated for a conceptual divertor design as the first step. Simulation of SOL/divertor plasmas is carried out at lower single null divertor (LSN) configuration with Ip/Bt=3.5 MA/2.5 T. For the present calculation, anticipated SOL power flux of Qtotal=35 MW and particle fuelling flux of Gion=5e21 /s (ne-edge=3e19 /m) are applied. The pumping speed (Spump=50 m3/s) is specified by an albedo for neutrals in front of the cryopump set bottom of exhaust chamber. The recycling of deuterium is assumed to be 100% at the first wall. For the first simulation, the carbon contamination in SOL/divertor regions is set to 2% of electron density uniformly. Gas puff flux Gpuff=0.5e21 /s is introduced from outside midplane. We assume particle diffusion coefficient D= 0.3 m2/s and thermal diffusivity of electron and ion Xe=Xi=1 m2/s. As a result, attached and detached plasma conditions are simulated on outer and inner divertor regions, respectively. The heat load around the outer strike point reaches 31 MW/m2, which largely exceeds the allowable range of 15 MW/m2 for CFC materials. Reduction of heat load must be achieved somehow. An effect of the radiation cooling is simulated to reduce such a large heat load as the second step. To enlarge the radiative cooling, we increased the gas puff flux by a factor of ten and the carbon contamination partly in the outer divertor region from 2% to 4%. It gives a favorable result that the peak heat load is reduced to 12 MW/m2 with radiation enhancement by a factor of two in the outer divertor region. Further study with wide parameter regions is being carried out with taking into account the pumping capability and with modified geometry for high heat and particle controllability. [1] M.Kikuchi, et al., Fusion Energy Conference (Proc.of 21th IAEA Conf.,2006), IAEA-CN-149/FT/2-5. [2] H. Kawashima, et al., Plasma Fusion Res. 1 (2006) 031.

CONCEPTUAL DESIGN OF A COMPONENT TEST FACILITY BASED ON THE SPHERICAL TOKAMAK

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A small steady state spherical tokamak (ST) offers an attractive system for producing simultaneously the neutron, particle and heat fluxes necessary to effectively test and optimise blanket modules, first wall structures and other components under the required fusion power plant conditions. This Component Test Facility (CTF) would complement and extend the qualification of materials by IFMIF and could operate in association with DEMO thus reducing the risk of delays, and extending the options, during this crucial stage of the development of commercial fusion power.
The ST-CTF offers many advantages including low tritium consumption, ease of maintenance and a compact assembly and would operate in a strongly driven mode in which Q~1. The current drive would be provided by a mix of bootstrap current and neutral beam injection systems. The blanket modules under test are removed and replaced using a casking system and the entire centre column assembly can be relatively easily removed, recycled and replaced via a hydraulic lift system beneath the tokamak assembly. The single turn toroidal field coil consists of a water-cooled copper centre rod with multiple return limbs, which offers a simple and robust structure requiring a low voltage, high current power supply. The poloidal field coils are also all water-cooled but use a glass fibre reinforced cyanate ester resin insulation that offers higher radiation resistance and higher strength than the conventional epoxy resin systems. When operated in H-mode most of the exhaust power is directed to the outer legs of the double null divertor configuration where high power densities and high material erosion rates are developed. A novel divertor target based on the use of a cascading flow of silicon carbide pebbles is being developed for this application.
This paper presents the current status of a conceptual design of an ST based CTF, its requirements and describes some of the technology issues and potential solutions that are being evaluated at Culham.

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DEPLETION ANALYSIS OF A SOLID TYPE BLANKET DESIGN FOR ITER

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For the blanket design in a D-T fusion power plant, the majority of the fusion energy is obtained from neutron kinetic energy, and the tritium to maintain self-sustainable D-T fusion reaction should be procured from the (n,t) reaction in the blanket. Thus neutronics analysis in the blanket design is indispensable. Especially for solid-type blanket design, neutronics analysis including depletion (transmutation) calculation is required to calculate more accurate neutron flux and tritium production rate. In the fission reactor design of an eigenvalue problem, the MONTEBURNS code is widely used for depletion calculations. However the fusion blanket design is a fixed source problem. In addition to this, the current version of the MONTEBURNS code does not involve Be9(n,t)Li7 and Li7(n,n't)alpha reactions. Thus, in this research, (1) the MONTEBURNS code is modified to solve the fusion blanket problem including Be9(n,t)Li7 and Li7(n,n't)alpha reactions and (2) a 3-dimensional neutronics depletion analysis of a solid type blanket for ITER is performed by using the modified MONTEBURNS code.

PS3-1009

NEUTRONICS ANALYSIS OF THE INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR (ITER) MCNP "BENCHMARK CAD MODEL" WITH THE ATTILA DISCRETE ORDINANCE CODE

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Extensive neutronics analyses are needed in ITER to ensure meeting the technical and functional requirements for various components design and to demonstrate the capability of safe operation. As such, neutronics calculations should conform to the ITER Management and Quality Program (MQP) that necessitates the availability of adequate design and modeling calculation tools which reduce turnaround time between CAD-based design changes and subsequent analyses. To ensure meeting QA requirements, ITER management has adopted a certain number of codes and data to perform the needed nuclear analyses. The MCNP Monte Carlo code is currently the reference tool along with the Fusion Evaluated Nuclear Data Library, FENDL2.1. Since it is envisioned that many design changes/modifications through close interaction with ITER CAD model designers will be required, there is currently extensive effort to develop and use a CAD-MCNP interface for design purposes to facilitate the modeling and analyses during this iterative process. This effort is in progress at the U. of Wisconsin (US), Forschungszentrum Karlsrohe (Germany) and the Institute of Plasma Physics (China). On the other hand, the ITER IT has adopted the newly developed FEM, 3-D, and CAD-based Discrete Ordinates code, ATTILA, as a potentially quicker alternative to the MCNP code for the neutronics studies contingent on its success in predicting key neutronics parameters and nuclear field according to the stringent QA requirements set forth by the MQP. ATTILA has the advantage of providing a full flux and response functions mapping everywhere in one run where components subjected to excessive radiation level and strong streaming paths can be identified. The ITER neutronics community had agreed to use a standard CAD model of ITER (40 degree sector, denoted "Benchmark CAD Model") to compare results for several responses selected for calculation benchmarking purposes to test the efficiency and accuracy of the CAD-MCNP approach developed by each party. Since ATTILA seems to lend itself as a powerful design tool with minimal turnaround time, it was decided to benchmark this model with ATTILA as well and compare the results to those obtained with the CAD MCNP calculations. In this paper we report such comparison for five responses, namely: (1) Neutron wall load on the surface of the 18 shield blanket module (SBM), (2) Neutron flux and nuclear heating rate in the divertor cassette, (3) nuclear heating rate in the winding pack of the inner leg of the TF coil, (4) Radial flux profile across dummy port plug and shield plug placed in the equatorial port, and (5) Flux at seven point locations situated behind the equatorial port plug. It should be mentioned that ATTILA has been recently benchmarked against the experimental results obtained from three fusion benchmark experiments performed at the FNG facility located in Frascati, Italy, and the results were encouraging. However, the results reported in this paper will be indicative of ATTILA's capability in performing calculations for large-scale and complex models such as ITER; and if meeting the QA requirements; it could be used as a complementary calculation tool for ITER nuclear design.

ATTACHMENT SYSTEM FOR DEMO IN-VESSEL COMPONENTS: BLANKET, MANIFOLD AND HOT RING SHIELD

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On the way to the first commercial nuclear fusion reactor (DEMO) new conceptual studies tackling the design of the in-vessel components and their integration into the reactor have been initiated. The main in-vessel components are the breeding blankets, the helium supply unit, namely the manifold, and the hot ring shield. For enabling the handling of these large components the in-vessel reactor components have to be segmented. With regards to this a new so called Multi-Module-Segment (MMS) concept has been developed, whereas blanket and manifold constitute vertical non-permanent segments to be installed and dismantled with remote handling tools through the upper port of the reactor. Each MMS consists of a number of blankets connected to a manifold block containing the helium cooling channels. The blankets which have to sustain the fuel cycle by breeding tritium and are also used to extract the produced heat are exposed to high thermal and mechanical as well as pressure loads. Therefore, the attachment system between blanket and manifold needs to be flexible to compensate different thermal expansion, but also needs to be stiff enough to withstand the loads during normal as well as non-normal operation, e.g. disruptions. The MMS itself are attached to the permanent hot ring shield (HRS) structure. Since between the hot ring shield and the vacuum vessel also temperature differences are existent the attachment system between MMS and HRS needs to be capable of compensating different thermal expansions. Additionally it needs to support the weight of the components and disruption loads, whereas remote handling is required for maintenance operations from inside the vacuum vessel.

The development of such an attachment system is a major engineering challenge, requiring reliability under harsh environmental and loading conditions. In this report requirements and proposals for possible solutions for the attachments between vacuum vessel and hot ring shield, between hot ring shield and MMS and between the manifold and blankets will be presented.

DEMO MAINTENANCE SCENARIOS: SCHEME FOR TIME ESTIMATIONS AND PRELIMINARY ESTIMATES FOR BLANKETS ARRANGED IN MULTI-MODULE-SEGMENTS

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Previous conceptual studies made clear that the ITER blanket concept and segmentation is not suitable for the environment of a potential fusion power plant (DEMO). One promising concept to be used instead is the so-called Multi-Module-Segment (MMS) concept. Each MMS consists of a number of blankets arranged on a strong back plate thus forming "banana" shaped in-board (IB) and out-board (OB) segments. With respect to port size, weight, or other limiting aspects the IB and OB MMS are segmented in toroidal direction. The number of segments to be replaced would be below 100.
For this segmentation concept a new maintenance scenario had to be worked out. The aim of this paper is to present a promising MMS maintenance scenario, a flexible scheme for time estimations under varying boundary conditions and preliminary time estimates.
According to the proposed scenario two upper, vertical arranged maintenance ports have to be opened for blanket maintenance on opposite sides of the tokamak. Both ports are central to a 180 degree sector and the MMS are removed and inserted through both ports. In-vessel machines are operating to transport the elements in toroidal direction and also to insert and attach the MMS to the shield. Outside the vessel the elements have to be transported between the tokamak and the hot cell to be refurbished.
Calculating the maintenance time for such a scenario is rather challenging due to the numerous parallel processes involved. For this reason a flexible, multi-level calculation scheme has been developed in which the operations are organized into three levels: At the lowest level the basic maintenance steps are determined. These are organized into maintenance sequences that take into account parallelisms in the system. Several maintenance sequences constitute the maintenance phases which correspond to a certain logistics scenario. By adding the required times of the maintenance phases the total maintenance time is obtained.
The paper presents preliminary time estimates for a conventional cask scenario, employing cranes and winches. The assumptions for the time estimates are based on industrial experience and especially on adapted results of ITER studies. Additionally, a corridor concept is treated taking advantage of the "two port scenario". Thus a permanent or semi-permanent installation of devices might be possible and the time for docking might drastically be reduced.

DEVELOPMENTS IN NUCLEAR LIQUID METAL TECHNOLOGY

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Current trends in the nuclear technology either in fission fusion or structure of matter research towards component solutions with high power densities led to a renaissance of the liquid metal related thermal-hydraulic and material research. Since liquid metals independent if they belong to the high or low Z-range offer the unique capability to act both as neutron source and as coolant they facilitate simple and robust structures within the reaction zone enabling compact designs at low capital investment. Due to their high specific electric and thermal conductivity they allow unique measurement and pumping techniques minimizing the effort for in-service inspection issues at simultaneously relatively moderate temperatures and temperature gradients within the structure. But, both the thermo-physical and the thermo-chemical properties of liquid metals require specific adapted solutions in order to match the individual goals.

In this context the KAlrsruhe Liquid metal LAboratory (KALLA) consisting of several stagnant and circulating liquid metal systems using both low and high Z- fluids has been erected and set into operation. KALLA is dedicated to investigate crucial thermal-hydraulic and material problems together with the development of adequate measurement and sensing techniques in the nuclear field. The aim of this article is to discuss significant developments conducted at KALLA supporting the research in the field of fusion. Moreover, it is aimed to overview the experiences gained with the operation of liquid metal facilities and to illustrate cross-cutting issues appearing not only in fusion research. The individual KALLA experimental facilities are now operated continuously since several years and a broad experience has been gained for components typically appearing in nuclear systems like pumps (both electromagnetic and mechanical), oxygen monitoring and control systems, etc.

VERIFICATION OF KERMA FACTOR FOR BERYLLIUM AT NEUTRON ENERGY OF 14 MEV BASED ON CHARGED-PARTICLE MEASUREMENT

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KERMA (Kinetic Energy Release in Materials) factors are essential data for calculation of nuclear heating, which is caused by recoiling nucleus or secondary emitted charged-particles via nuclear reactions. They are deduced from evaluated nuclear data libraries and are used in fusion nuclear design.

Previously direct measurements of nuclear heating for various materials induced with DT-neutrons were conducted at the Fusion Neutronics Source (FNS) facility in JAERI. It was pointed out that measured heating for beryllium disagreed with calculated ones with KERMA. The calculated heating value with JENDL-3.2 underestimated the measured one by 25 %. Reasons of the discrepancy have not been understood clearly.

Recently we measured the alpha-particle emission cross-section for beryllium with DT-neutron incidence. From the measurement, it was found that JENDL-3.3 underestimated cross-section of high energy alpha-particle emission at forward angles, while JEFF-3.1 showed rather well agreement. This result suggested the evaluation of Be-9(n, 2n+2alpha) reaction in JENDL had some problems and offered a possibility of underestimation of the KERMA factor of beryllium.

In order to estimate the KERMA factor, we proposed a model of the Be-9(n, 2n+2alpha) reaction according to our experimental results. Then, a new KERMA factor was calculated by the direct method based on our experimental model. It was also deduced from the latest nuclear data libraries, JENDL-3.3, ENDF/B-VI and JEFF-3.1 with NJOY99. The KERMA factor from JENDL underestimated our experiment-based one by 15 %, while the other ones rather agreed with ours. This result was consistent with the previous direct measurement of nuclear heating and a specific problem of the JENDL evaluation emerged. The superiority of the latest JEFF was demonstrated through the present comparison.

This study would be a typical example that detailed measurements of differential cross-sections for emitted charged-particles and investigation of a reaction model are useful for evaluation and verification of KERMA factors of light nuclei.

SIMULATION OF PLASMA PARAMETERS FOR HCSB-DEMO BY 1.5D PLASMA TRANSPORT CODE

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The main goals of the paper are aimed at simulating core plasma parameters of HCSB-DEMO (Heilum-cooled Solid Breeder, HCSB) by 1.5D plasma transport code. The study content included: the operation scenarios; the temperature and density profiles of the ion and electron; fusion and radiation power; the distribution of current density and safety factor; sensitivity analyses for some of the input parameters and physical models parameters, finally, there is a primary estimate of the divertor¡¯s target loads. The fusion power output of 2.6GW with a major radius of 7.2m, aspect ratio of 3.4, elongation of 1.85, triangularity of 0.45, plasma current of 14.8MA, normalized beta of 4.4, maximum field of 13T, electron density of 1.5¡Á1020/m3, average electron temperature of 14.5keV and neutron wall loading of 2.3 MW/m2. In this investigation, parameters of reactor which satisfy the DEMO requirement are selected.

Keyword: simulation; plasma parameter; HCSB-DEMO

COMPARISON AND ANALYSIS OF 1D/2D/3D NEUTRONICS MODELING FOR A FUSION REACTOR

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During the course of analyzing the characteristics for fusion reactors, the refined calculations are needed to confirm that the nuclear design requirements are met. Since the long computational time is consumed, the refined three-dimensional (3D) representation has been used primarily for establishing the baseline reference values, analyzing problems which cannot be reduced by symmetry considerations to lower dimensions, or where a high level of accuracy is desired locally. The two-dimensional (2D) or one-dimensional (1D) description leads itself readily to resolve many problems, such as the studies for the material fraction optimization, or for the blanket size optimization. The purpose of this paper is to find out the differences among different geometric descriptions, which can guide the way to approximate and simplify the computational model.

The fusion power reactor named FDS-II was designed as an advanced fusion power reactor to demonstrate and validate the commercialization of fusion power by Institute of Plasma Physics, Chinese Academy of Science. In this contribution, the dual-cooled lithium lead (DLL) blanket of FDS-II was used as a reference for neutronics comparisons and analyses. The geometric descriptions include 1D concentric sphere model, 1D, 2D and 3D cylinder models. The home-developed multi-functional neutronics analysis code system VisualBUS, the Monte Carlo transport code MCNP and nuclear data library HENDL have been used for these analyses.

The neutron wall loading distribution, tritium breeding ratio (TBR) and nuclear heat were calculated to evaluate the nuclear performance. The 3D calculation has been used as a comparison reference because it has the least errors in the treatment of geometry. It is suggested that the value of TBR calculated by the 1D approach should be greater than 1.3 to satisfy the practical need of tritium self-sufficiency. The distribution of nuclear heat based on the 2D and 3D models were similar since they all consider the effects of the axial components. The differences between the results with the MC and SN method were also presented. Comparison of the results suggested that, for obtaining global scalar quantities for general use, simplified models of the fusion reactor are usually sufficient. However, more calculations are needed to test the validity of each model in different regions of phase space.

Key words: fusion, neutronics, model, FDS-II

CONCEPTUAL DESIGN OF CHINA FUSION POWER PLANT FDS-II

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As one of the series of fusion system design concepts developed by the FDS Team of China, FDS-II is designated to exploit and evaluate potential attractiveness of fusion energy application for the generation of electricity on the basis of conservatively advanced plasma parameters, which can be limitedly extrapolated from the successful operation of ITER. The principle of the blanket design is established in both the feasibility and potential attractiveness of technology to meet the requirement for tritium self-sufficiency, safety margin, operation economy and environment protection etc.

The plasma physics and engineering parameters of FDS-II are selected on the basis of the progress in recent experiments and associated theoretical studies of magnetic confinement fusion plasma with a fusion power of 2~3GW. The neutron wall load of 2~3MW/m2 and the surface heat flux of 0.5~1MW/m2 are considered for high effective power conversion.

The "multi-modules" scenario is adopted in the FDS-II blanket design to reduce thermal stress and electromagnetic forces under plasma disruption, with liquid metal lithium lead (LiPb) as tritium breeder, the Reduced Activation Ferritic/Martensitic (RAFM) steel as structural material. Two options of specific liquid LiPb blanket concepts have been proposed, named the Dual-cooled Lithium Lead (DLL) breeder blanket and the Quasi-Static Lithium Lead (SLL) breeder blanket.

The DLL blanket is a dual-cooled LiPb breeder system with helium gas to cool the first wall and main structure and LiPb eutectic to be self-cooled. The flow channel inserts (FCIs), e.g. SiCf/SiC composites, are designed as the thermal and electrical insulators inside the LiPb flow channels to reduce the magnetohydrodynamic (MHD) pressure drop and to allow the coolant LiPb outlet temperature up to 700 C for high thermal efficiency.

The SLL blanket is another option of the FDS-II blanket with the technology developed relatively easily. To avoid or mitigate the problems resulting from MHD effects and FCI technology and the corrosion from the high temperature LiPb, the SLL blanket is designed to use quasi-static LiPb flow instead of fast moving LiPb with lower LiPb outlet temperature. The heat in the SLL blanket is removed by pressurized helium gas with the outlet temperature of ~450 C.

This paper gives an overview of the FDS-II conceptual design covering plasma physics and engineering, blanket neutronics and thermalhydraulics, safety and environmental impact, cost and benefit analyses etc, including systematic comparison analyses between the two kinds of blankets. The further R&D needs are specified as well as the existing basis of R&D in China.

Key words: Fusion power plant, Conceptual design, Liquid blanket, LiPb breeder

OPTIMIZATION ACTIVITIES ON DESIGN STUDIES OF LHD-TYPE REACTOR FFHR

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An overview of recent activities on optimizing the base design of LHD-type helical power reactor FFHR is presented, including key design improvements on blanket and divertor configurations, neutronics performance, large superconducting coils and cost models. New proposals of plasma operation candidates are also shown.
In the reference design FFHR2m1 with the major radius R of 14 m, one of the main issues is the structural compatibility between blanket and divertor configurations. In particular, the blanket space at the inboard side is still insufficient due to the interference between the first walls and the ergodic layers surrounding the last closed flux surface. To overcome this problem, helical x-point divertor (HXD) has been proposed to remove the interference. In this concept, very effective screening of recycling neutrals with 99% ionization is expected according to 3D simulations. Another approach for obtaining larger clearance is also being examined by modifying the configurations of the helical coils. Neutronics performances on liquid breeder blanket have been also improved with optimizing the neutron multiplier to achieve the local TBR over 1.3 and a sufficient shielding efficiency of the fast neutron fluence of lower than 1E22 n/m2 in 30 years. This issue on the blanket configurations also includes the issue on net cover rate of the inner wall for the total TBR over 1.2. For this requirement, the discrete pumping with semi-closed shield (DPSS) is proposed to achieve the net cover rate of over 0.9, which is very advantageous to suppress nuclear streaming. The other key issues are the engineering aspects on large superconducting coils. Poloidal coils positions are optimized to be compatible with 3D configurations of blankets within an acceptable total magnetic energy. Promising candidates of R&W (react and winding) of CICC (cable in conduit conductor), indirect cooling magnets with external dumping for quenching, and the LHD-type support posts for the total 16,000 ton of the cold mass and the maximum 55 mm thermal deformation are preliminary proposed. Magnetic field perturbation using poloidal field coils is newly proposed to selectively remove cold alpha-particles. To evaluate those optimization and improvements, a new cost model is proposed to make clear parameter sensitivities. Current-less helical plasma gives great advantages on plasma operations. The external heating power is needed only at the ignition access phase, and the minimum power is found to be less than 30 MW at a longer startup than 300 sec. This flexibility contributes in reducing the total areas of ports occupied by heating devices. High-density ignition with lower temperature is also promising using superdense core (SDC) plasmas recently discovered in LHD. Regarding this new ignition access scenario, comprehensive control methods of thermal instability are discussed.

MACHINE SIZE REDUCTION EFFECT AND FEASIBILITY OUTLOOK FOR CS-FREE TOKAMAK REACTOR

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A slimming down or discard of the CS (center solenoid) coil leads to a tokamak machine significant reduction in weight. The effects are quantitatively evaluated. Major effect appears on a TF (toroidal field) coil weight, and is caused by an aspect ratio lowering. This effect can be called ST (spherical torus) effect. The aspect ratio lowering opens up high plasma performances e.g. higher elongation and higher beta. Therefore the requirement for the toroidal field strength is mitigated. Another effect of the aspect ratio lowering on the reactor structural configuration is a wide open area between the adjacent TF coils with a help of the TF coil slimming down. The wide open area allows us to choose a so-called Hot cell maintenance approach where the sectors are removed from the power core to the hot cell for refurbishment.

On the other hand, the departure from the power transformer causes the plasma current control difficult. Focusing on the CS-free tokamak plasma, the plasma initiation (break down), current ramp up, current sustain and current ramp down are simulated with consideration of the plasma parameters (temperature, density and current) profile effects. As the non-inductive external current drive devices, a NBI (neutral beam injection) system or an ECW (electron cyclotron wave) system are adopted. The current drive efficiency of the NBI system is twice higher than the ECW system. In the meantime, the ECW system can be arranged without adversely affecting the maintenance (replacement of torus sectors) performance. This may be more essential than the drive efficiency for the power reactor. The current ramp up time is estimated as about one hour both for these drivers. Even for the lower efficient driver i.e. the ECW system, the fusion gain Q is higher than 25 when a bootstrap current fraction is higher than 80%. The fusion gain Q of 25 seems to be acceptable level for the power plant.

MAINTENANCE APPROACH OF FINAL OPTICAL DEVICES FOR A FAST IGNITION ICF REACTOR

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A fast ignition inertial confinement fusion power plant concept with a dry first-wall and a high repetition laser, which is based on a possibility of fast ignition method to achieve the large energy gain with lower laser power, has been proposed. The replacement procedure of in-vessel and ex-vessel components is one of key issue for operation of an ICF power plant. It is considered that maintenance of ICF reactor components is simpler than that of MCF reactor, because of complicated arrangement of super-conducting coils around the core of MCF reactor. However, that of ICF reactor is not so simple taking the complicated arrangement of numbers of beam lines and final optical devices into consideration. The authors have already discussed the maintenance approach of the tritium breeding blanket of a fast ignition ICF reactor. In the present study, the arrangement final optical devices, which are supposed to be exposed directly to the high energy neutron flux from the reactor core, and their maintenance approach are discussed.
The proposed ICF reactor concept has thirty-one compression laser beam lines. Every beam line has a cartridge of final optical devices, which consists of a transmissive diffraction grating and some controllable reflection mirrors. In order to reduce the neutron streaming, the beam line is bended several times in the cartridge by controllable mirrors. The cartridges are inserted into the chamber-side of maintenance caves penetrated in the wall of a cylinder-shaped chamber room, which plays a roll of shielding. A vacuum vessel, in which blanket sectors with dry first-wall are settled, is placed at the center of the chamber room. The vacuum vessel plays a roll of tritium boundary.
Procedure of the replacement of the final optical devices is as follows;
1) Shielding blocks in the rear-side of the maintenance cave are removed from outside of the wall. 2) A cask-type replacement device is transferred to the outside of the maintenance cave on a corridor. Six access corridors are placed along the outside of the wall. 3) A cartridge handling device installed in the cask draws out the cartridge through the maintenance cave. 4) The cartridge taken out is transported to the hot cell zone with the cask-type replacement device. 5) A new cartridge is transported into the cave by a cask-type replacement device. 6) The shielding blocks is filled in the rear-side of the maintenance cave.
The cartridge of final optical devices is placed at outside the tritium boundary for the reactor core. However, as the inside of the cartridge is exposed to the tritium atmosphere, A shutter of laser beam line, gas exchange system in the cartridge and cask-type replacement device are required to prevent the tritium release during the replacement.

INVESTIGATION OF CASCADE-TYPED FALLING LIQUID FILM FLOW ALONG FIRST WALL OF LASER-FUSION REACTOR

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To protect from high energy/particle fluxes caused by nuclear fusion reaction such as extremely high heat flux, X rays, Alpha particles and fuel debris to a first wall of an inertia fusion reactor, a "cascade-typed" falling liquid film flow is proposed as the "liquid wall" concept which is one of the reactor chamber cooling and wall protection schemes: the reactor chamber can protect by using a liquid metal film flow (such as Li_17Pb_83) over the wall. In order to investigate the feasibility of this concept, we conducted the numerical analyses by using the commercial code (STREAM: unsteady three-dimensional general purpose thermofluid code) and also conducted the flow visualization experiments. The numerical results suggested that the cascade structure design should be improved, so that we redesigned the cascade-typed first wall and performed the flow visualization as a POP (proof-of-principle) experiment.
In the numerical analyses, the water is used as the working liquid and an acrylic plate as the wall. These selections are based on two reasons: (1) from the non-dimensional analysis approach, the Weber number (We=ru^2d/s: r is density, u is velocity, d is film thickness, s is surface tension coefficient) should be the same between the design (Li_17Pb_83 flow) and the model experiment (water flow) because of the free-surface instability, (2) the SiC/SiC composite would be used as the wall material, so that the wall may have the less wettability: the acrylic plate has the similar feature.
The redesigned cascade-typed first wall for one step (30 cm height corresponding to 4 Hz laser duration) consists of a liquid tank having a free-surface for keeping the constant water-head located at the backside of the first wall, and connects to a slit which is composed of two plates: one plate is the first wall, and the other is maintaining the liquid level. This design solved the trouble of the previous design. The test section for the flow visualization has the same structure and the same height as the reactor design. The test section consists of three steps of the cascade-typed first wall, and the water is supplied to the tanks on the top and middle steps of the test section and then it makes liquid film flow on the first wall. When the water flow rate becomes over the Weber number coincident with the reactor design, the liquid film is stably flowing on the first wall for each step. On the other hand, when the liquid flow rate decreases less than the above Weber number condition, the liquid film flow divides into two or more streams due to the wettability of the wall. However, since we can control the liquid flow rate, the thickness of the liquid film can be controlled, too. This suggests that it can control the liquid film flow velocity under the reactor condition. Further analyses and the experiments are conducting now and we will report it in the final paper.

IMPLEMENTATION OF GAS TARGET ON THE LIL FACILITY

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The Laser MegaJoule (LMJ) is a new facility under construction close to Bordeaux (France) which will be available in the 2010's in France to study Inertial Confinement Fusion. This stadium-size facility will be able to deliver 1.8MJ of UV light in few nanoseconds. The Ligne d'Integration Laser (LIL) is the prototype of an LMJ bundle (a gathering of 8 laser beams). Among this, 4 beams (a quad) are routinely operational to perform laser-matter interaction experiments.

This facility includes a target bay where the UV laser from the quad is focused. The LIL target is a complex assembly of several mechanical elements with a specific goal for each of them. These are needed to perform the experiment (main target, hohlraum for example), to backlight the main target (X-ray radiography sources), to hold the elements (assembly with LIL mechanical interface), to align the target (specific patterns glued on specific carrier) and to ensure gas conditioning.

The purpose of this poster is to present how gas-containing targets are implemented on the LIL facility.

The main goal of gas target conditioning is to provide, for post-shot analysis, a precise measurement of the pressure and the right composition of gas in the millimeter-size hohlraum, as close as possible (typically a minute before) from the shot. To do this, specific equipments have been developed to vacuum test and fill the targets (with a monitored vacuum tank coupled with Helium spectrometer), before delivering them to the facility.

At the same time, an integrated system was developed, which is part of the target positioner. Until it is switched off a few seconds before the laser shot, it is able to measure regularly the gas pressure and to compare it to the requested value while the target is ready at the target chamber center. An evolution of this system is prepared now, to make it able to adjust the gas pressure, with a feedback loop, to match the request with an enhanced accuracy.

SATURATED MAGNETIC FIELDS OF WEIBEL INSTABILITIES IN ULTRAINTENSE LASER-PLASMA INTERACTIONS

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Recent advancements in ultraintense, short pulse lasers have allowed for the exploration of many novel regimes in the field of laser-plasma interactions.Energetic charged particles generated by laser-plasma interactions can be used in many applications including the initiation of tabletop particle accelerators as well as fusion by fast ignition. The generation of energetic particles by the interaction of an ultraintense laser pulse with a plasma has been demonstrated in theoretical and experimental studies. A fast ignitor concept was proposed as the approach to efficiently ignite the high density fusion fuel plasmas with an ultraintense short pulse laser. In the fast ignitor scheme, the intense laser pulse propagates through a colonal plasma up to several times the critical density and delivers energy to high energy electrons. These highly energetic particles then transport the energy through the overdense plasma to the center of the compressed core and ignite the fuel there. It is known that interactions of ultraintense laser pulses with overdense plasmas lead to the generation of a magnetic field (the so called Weibel instability). The Weibel instability breaks up the high energy electron current into filaments. It is known that the self-generated magnetic fields play a crucial role in this energytransport. The self-generated magnetic fields are a serious obstacle to the realization of the fast ignitor scheme. There have been a number of many reportson the Weibel instability of self-generated magnetic fields. It is very importantto determine the saturated magnetic fields. The interaction of ultraintense laser pulses with overdense plasmas is studied by theory as well as three-dimensional particle-in-cell simulations. Self-generated magnetic fields are observed in the plasma target owing to the Weibel instability. The growth ratesof the self-generated magnetic fields and the saturated magnetic fields in our theory are in good agreement with the results of our simulations. It is found that the saturated magnetic fields of Weibel instabilities are determined by thelaser intensity and the plasma density.

LASER FUSION REACTOR DESIGN IN A FAST IGNITION WITH A DRY WALL CHAMBER

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One of the critical issues in laser fusion reactor design is high pulse heat load on the first wall by the X-rays and the fast/debris ions from fusion burn. There are mainly two concepts for the first wall of laser fusion reactor, a dry wall and a liquid metal wall. We should notice that the fast ignition method can achieve sufficiently high pellet gain with smaller (about 1/10 of the conventional central ignition method) input energy. To take advantage of this property, the design of a laser fusion reactor with a small size dry wall chamber may become possible.
Since a small fusion pulse leads to a small electric power, high repetition of laser irradiation is required to keep sufficient electric power.
Then we tried to design a laser fusion reactor with a dry wall chamber and a high repetition laser. This is a new challenging path to realize a laser fusion plant.
Based on the point model of the core plasma, we have estimated that fusion energy in one pulse can be reduced to be 40 MJ with a pellet gain around G>100. To evaluate the validity of this simple estimation and to optimize the pellet design and the pulse shaping for the fast ignition scenario, we have introduced 1-D hydrodynamic simulation code ILESTA-1D and carried out implosion simulations. Since the code is one-dimensional, the detailed physics process of fast heating cannot be reproduced. Thus the fast heating is reflected in the code as the additional artificial heating source in the energy equation. It is modeled as a homogeneous heating of electrons in core region at the time just before when the maximum compression is achieved. At present we obtained the pellet gain G~100 with the same input energy as the above estimation by a simple point model (350kJ for implosion, 50kJ for heating and assuming 20% coupling of heating laser).
A dry wall is exposed to several threats due to the cyclic load by the high energy X-ray and charged particles: surface melting, physical and chemical sputtering, blistering and exfoliation by helium retention, and thermo-mechanical fatigue. Here we have developed 1-D thermal analysis code, where the energy spectra calculated in the pellet implosion simulation has been incorporated into the thermal analysis code as the thermal load to the first wall. According to 1-D thermal analysis, a dry wall chamber of R=5.64m is feasible from the viewpoint of temperature evolution under this condition with the first wall of tungsten-armed ferritic steel.

SC DOPED CAZRO3 HYDROGEN SENSOR FOR LIQUID BLANKET SYSTEM

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The chemical control is one of the critical issues for self-cooled liquid breeder blanket system of fusion reactors. Especially, hydrogen level is the key parameter for corrosion and mechanical properties of the in-reactor components. In addition, the control of tritium is essential for the performance of breeding blankets. Therefore, on-line hydrogen (isotopes) sensing is an essential technology. In the present study, an on-line hydrogen sensor made of proton conducting ceramics was developed, and the performance of the sensor was investigated in the expected conditions for liquid lithium (Li) and LiF-BeF2 (Flibe) blankets. The evaluation of the predicted performance of the sensor in Li and Flibe was carried out by means of the performance test in gas atmosphere and liquid metals at hydrogen partial pressures equivalent to those for the blanket conditions.

In the present work, Sc doped CaZrO3 (CaZr0.95O3-aSc0.05) sensor was developed for liquid blanket systems. The design was based on In doped CaZrO3 (CaZr0.9O3-aIn0.1) sensor for molten aluminum [1]. The Sc doped CaZrO3 is thermodynamically stable more than the In doped one in the reducible melt, such as Li. The sensor is a cap-shaped one, which has a gas compartment. The cell is immersed into the melt which contains hydrogen at the partial pressure of PH1. Then, the hydrogen partial pressure of PH1 in the melt becomes equilibrium with that in the gas compartment according to the Sievert law. The reference cell is filled with reference gas at hydrogen partial pressure of PH2. Then, the electromotive force (EMF) is obtained by the difference of the hydrogen partial pressure in the electro chemical system of PH1 (melt) | solid electrolyte | PH2 (reference cell). The Nernst equation is used for the evaluation of the hydrogen partial pressure from the obtained EMF. For the measurement of the hydrogen in the melt using the proton conducting ceramics, the partial pressure of oxygen in the melt is important. This is because proton permeation is caused by the proton capture in the hole, which is formed when the oxygen is captured in an oxygen ion vacancy in the solid electrolyte [2].

The sensor performance tests were carried out in Ar-hydrogen gas atmosphere, molten Al and liquid Li at the hydrogen partial pressures equivalent to those for the melts in the reactor conditions. The hydrogen partial pressure in the gas varied from 10-14atm to 10-3atm. In the gas atmosphere test at the temperature of 773K, 873K and 973K, the sensor showed quick response with reproducibility to the applied change in hydrogen partial pressure. The performance and durability of the sensor in the melt of Al at 973K and Li at 873K were also investigated.

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CONSTITUTIVE EQUATIONS OF LI2TIO3 AND LI4SIO4 PEBBLE BEDS OBTAINED BY MEANS OF STANDARD TRIAXIAL TESTS : IMPLEMENTATION OF THE MODEL IN A FEM CODE

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During the ITER operation, some Test Breeding Modules (TBM) will be tested in the equatorial ports of the reactor. Presently, these TBM are being developed in different research centres in the world. The Helium-Cooled Pebble Bed (HCPB) blanket is one of two breeder blanket concepts developed in Europe. The HCPB uses Beryllium pebbles as neutron multiplier and. Li4SiO4 or Li2TiO3 pebbles as breeder material.

The analysis of the breeding blanket is complex for the geometry as well as for the loading conditions. Advanced computer codes are needed in order to determine the reliability of the different designs. In this context the mechanical characterisation of pebble beds is important in order to simulate their behaviour. This paper illustrates the standard tests, performed in order to obtain the effective properties of the pebble beds, and the implementation of a constitutive model of the granular material in a FEM code. Several Authors have analyzed the pebble bed by meand uniaxial compression tests (oedometer tests). This test permits to obtain an effective displacement- load law under lateral constraint, but no data are obtained about the pebble bed shear resistance or about the three-dimensional behaviour of the bed. In the soil (made of sand, gravel or clay) qualification, triaxial tests are used for determining all their constitutive properties. In these test the soil is loaded by axial and lateral loads which can be varied independently. The measurement of the load and the displacement in both the directions permits to obtain the material constants f the constitutive models elaborated for describing the soil behaviour. The classic soil models are the Cam-Clay model and the Drucker-Prager with cap model. These models are implemented in several commercial FEM codes and they could be easily used for simulating the pebble beds. But the pebble bed behaviour is different from that of the soil. The soil models describe in detail the behaviour dependent on the water pressure and on the drainage conditions. These aspects have not any meaning for the pebble bed. Moreover the soil consolidation is different from the creep of the pebble bed. The paper demonstrates the limits of applying the soil model to the pebble bed. In fact the triaxial tests have been simulated numerically by means a commercial FEM code considering the classic soil models and the material constants obtained by the tests. Moreover the paper emphasizes the relative importance of the material constants (about 10), contained in the classic soil model, in order to fit the experimental results of the tests on the pebble beds.

EXPERIMENTAL TESTS AND THERMO-MECHANICAL ANALYSES ON THE HEXCALIBER MOCK-UP

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Within the frame of the R&D activities promoted by European Fusion Development Agreement (EFDA) on the Helium Cooled Pebble Bed Test Blanket Module (HCPB-TBM) to be tested and qualified in ITER, ENEA and the Department of Nuclear Engineering of the University of Palermo (DIN) have been fruitfully cooperating since several years to investigate the thermo-mechanical constitutive behaviour of both Beryllium and lithiated ceramics pebble beds, by adopting both an experimental and a theoretical approach. In particular, ENEA has carried out several experimental campaigns on small scale mock-ups tested in out-of-pile conditions (TAZZA, HELICHETTA and HELICA), while DIN has developed a proper constitutive model for the prediction of the pebble bed thermo-mechanical performances by using a Finite Element Method (FEM) commercial code. EFDA has more recently proposed to assess, by a benchmark exercise among ENEA-DIN, FZK and NRG, which constitutive model and relevant FEM code could be adopted as design tool for HCPB-TBM. The benchmark and the relevant comparisons are being performed using the results from ENEA Brasimone experimental test campaigns on HELICA and HEXCALIBER mock-ups. After having concluded the tests on HELICA, ENEA is preparing the tests on HEXCALIBER to investigate the thermo-mechanical behaviour of both Beryllium and Lithium Orthosilicate pebble beds when mutually interacting in adjacent cells and reactor-relevant geometries. HEXCALIBER was designed and manufactured to reproduce a portion of the former HCPB-TBM with two Lithium Orthosilicate and two Beryllium pebble bed cells both heated by couples of flat electrical heaters. The mock-up will be tested in HE-FUS 3 facility, under adequate adjustment of bed temperatures, temperature gradients, coolant temperatures, flow distributions and mechanical constraints, to assess the thermo-mechanical performance of the pebble beds under steady state and cyclic heat power conditions. The first test campaign is planned to be performed in the first part of 2007 using Lithium Orthosilicate (0.24-0.40 mm in diameter) pebbles as breeder material and Beryllium (1 mm in diameter) pebbles recently qualified as neutron multiplier. Among other experimental results, the temperatures, both in lithiated ceramics and Beryllium beds, and the overall displacements of the box will provide useful data to be compared with the theoretical ones computed by FEM codes with adequate models for pebble beds. The paper presents the main features of the HEXCALIBER mock-up, the detailed description of its experimental set-up and the first results of the experimental campaign. Moreover, the numerical simulation carried out by DIN on a 3D FEM model of HEXCALIBER, in the frame of the benchmark exercise, is also presented with details on the mock-up and cooling system nodalization and adopted loads and boundary conditions. Finally, the most significant numerical results, concerning temperatures and displacements at the instrumented points of the mock-up, are critically compared with the relevant measured values.

MANUFACTURE OF A SHIELD PROTOTYPE FOR PRIMARY WALL MODULES

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In the frame of the BLANKET MODULE (BM) development for ITER, an R&D programme was implemented for the manufacture of a shield prototype by powder Hot Isostatic Pressing (HIPping).
The manufactured shield is a full scale module N° 11a. Starting from a forged block of 1200 * 1200 * 500 mm, the main machining steps as deep drilling (1200 mm), 3D machining and sawing were performed. Tubes were 3D bent and large number of small parts were designed and machined. By welding together all the sub-parts we erected the main part of the water coolant circuit. Once the water circuit was built; the shield was completed using powder HIPping together with forged block embedding the tubes and their in a final solid part. The powder / solid HIP is used to minimize the number of BM seal welds in front of plasma. It increases the reliability of the components during operation.
About 300 kg of stainless steel powder was densified together with the forged block.
3D measurement was done before and after the HIP cycle to collect the data to be compared with theoretical model. It allows to predict the main distortions of the solid bulk.
Ultrasonic examination of the densified powder on the Stainless steel bulk and around the bended tubes was performed as well as mechanical characterization of the samples.
The recess for stub key attachment on the vacuum vessel side, the hydraulic connector, the key for the primary wall panel attachment on the front side and the link between the four parallel water coolant circuits were then machined to achieve the shield prototype.

PS3-3010

EXAMINATION OF ELECTRICAL INSULATING PERFORMANCE OF ER2O3 CERAMIC COATING UNDER ION BEAM IRRADIATION

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Electrical insulating ceramic coating has been developed for reducing MHD pressure drop in the Li/V-alloy blanket system. Recent studies indicate that a coating of Er2O3 is chemically stable in highly corrosive high-temperature liquid Li environment and that the Er2O3 coating on vanadium alloys has a potentiality for self-healing. In the present study, influence of radiation damage on insulating performance of Er2O3 coating has been examined by ion beam irradiation.
Coating layers of Er2O3 have been deposited on polished stainless steel plates of 15 x 15 x 2 mm3 with RF sputtering. The thicknesses of the Er2O3 layers were 2.0 microns (Sample A) and 2.4 microns (Sample B). For examination of the electrical properties, platinum electrodes of 2 x 2 mm2 were made on the insulating layers with DC sputtering. Before ion beam irradiation, the initial conductivities of Samples A and B were 1.7 x 10-12 S/m and 7.4 x 10-12 S/m, respectively. Irradiations of 1.7 MeV H+ and 2.8 MeV He+ beam on the coating layers have been performed at Institute for Materials Research, Tohoku University. The ion beam was injected perpendicular to the coating layers through the thin platinum electrodes of 150 nm in thickness. The beam irradiation was stopped at programmed ion fluences and the conductivities were measured. Changes in the electrical properties due to damage were examined by repeating the beam irradiations and the conductivity measurements at room temperature. The estimated maximum irradiation damages on Samples A and B were 0.24 and 0.025 dpa, respectively.
Significant change in the electrical conductivity has not been observed for Sample A during the irradiations of up to 0.24 dpa. During the ion irradiations on Sample A, no bias voltage was applied to the coating layer up to ~0.1 dpa. From ~0.1 dpa to 0.24 dpa, the bias voltage of 0.2 V, which is corresponding to the strong electric field of 1 kV/mm estimated for the blanket condition, was applied during the beam irradiations. On the other hand, the electrical conductivity of Sample B increased gradually with the irradiations in the low damage region of 10-3 - 10-2 dpa, while no bias voltage was applied during the beam irradiations. The conductivity was 5.0 x 10-10 S/m at 0.025 dpa. To obtain information on the mechanism of the degradation in insulating performance, the RIC (radiation induced conductivity) of Sample B was evaluated from the slope of I-V (current-voltage) curve under the beam irradiations. Although the electrical conductivity, measured during interval of the beam irradiation, dramatically increased by two orders of magnitude with the damage of up to 0.025 dpa, significant change in the RIC has not been observed. The present results indicates the possibility of degradation of the insulting performance due to a leakage current path locally produced by the ion irradiations even at low irradiation fluence.

RADIATION DAMAGE EFFECT ON THE PERFORMANCE OF TRITIUM PERMEATION BARRIERS

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To provide a means for the active control of hydrogen isotope permeation into and through metallic components of fusion reactor systems, thin ceramic coatings have been proposed. One of the important aspects of this barrier application in fusion is the stability of its permeation properties under neutron irradiation. So far, there is no data on how these barriers perform after neutron irradiation. To simulate neutron damage we exposed samples to ion beam irradiation in our 3 MeV tandem accelerator facility.
The barriers selected were aluminium oxide and erbium oxide films with thicknesses up to 1 µm, which had earlier been proved to suppress deuterium permeability from the gas phase by a factor of up to 1000. They were deposited on EUROFER substrates by the filtered vacuum arc deposition technique. Defect production and the estimation of the achieved displacements per atom (dpa) were modelled using the SRIM2006 computer code. The beam energy was varied during the irradiation in order to improve the uniformity of the damage profile within the oxide film. An average defect production of 5 to 10 dpa was obtained. To obtain lateral uniformity of the damage pattern along the sample surface, a beam-sweep system was employed.
Permeation tests of non-irradiated and irradiated samples were conducted. The former one showed a permeation reduction factor (PRF) similar to that reported earlier. As compared to non-irradiated, the irradiated samples revealed values of PRF with a moderate decrease of performance upon irradiation. Furthermore, it was observed that the lag time for occurrence of permeation for irradiated samples is much higher than for the pristine sample. A thorough analysis of the samples before and after permeation tests was carried out to reveal how the irradiation damages could affect the process of hydrogen transport in ceramic barrier coatings.

IMPACT OF REFLECTED NEUTRONS ON PREDICTION ACCURACY OF TRITIUM PRODUCTION RATE IN FUSION REACTOR

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A series of breeding blanket neutronics experiments with DT neutrons for Japanese ITER test blanket module have been carried out at JAEA FNS and tritium production rates (TPR) inside blanket mockups were measured in details. The measured TPRs were compared with calculations with the nuclear data libraries (FENDL-2.1 and JENDL-3.3) and the Monte Carlo code
MCNP4C in order to estimate prediction accuracy for TPR. The calculations agreed with the measured TPRs well except for the following two cases.
1. Breeding blanket experiment with a reflector : The blanket mockup was composed of 6Li 40% enriched Li2TiO3 breeder, beryllium layers. DT neutron source was surrounded by a reflector of SS316, which produced reflected neutrons incident to the blanket mockup. In this experiment the calculation overestimated the measured TPR by 10% clearly.
2. Pebble bed blanket experiment : The mockup was composed of 15 mm thick Li2O (natural enrichment) pebble bed layer (pebble diameter : 1 mm), 101.6 mm thick beryllium block and 1.8 mm thick F82H container. The calculation agreed with the measured TPRs well around the front boundary between the breeder layer and the container, while it overestimated up to around 10 % with depth in the breeder layer around the rear boundary.
The above results suggested that the calculation could not well represent reflected neutrons from the reflector and beryllium well. As reflected neutrons are always produced in real fusion reactors, it is a serious problem that they worsen prediction accuracy of TPR. It was inferred that the discrepancy between the measurement and calculation was due to inadequate angular distributions, particularly backward part, in the nuclear libraries of iron, beryllium, etc. because there were few experimental data on double-differential cross sections to backward directions. In order to confirm our surmise, we tentatively decreased backward part of angular distributions of iron, beryllium, etc. in FENDL-2.1 and JENDL-3.3 and investigated the effects. As a result, discrepancy between the measurement and calculation decreased in the above two cases.Angular distributions in the nuclear data libraries should be revised based on this result in order to increase prediction accuracy of TPR in fusion reactors.

DNS AND K-EPSILON MODEL SIMULATION OF MHD TURBULENT CHANNEL FLOWS WITH HEAT TRANSFER

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Molten salt FLiBe is one of the coolant candidates in the first wall and blanket of the fusion reactors and has the advantages in MHD (Magneto-Hydro-Dynamics) pressure loss, chemical stability, solubility of tritium and so on. In the contrast, the low thermal diffusivity and high viscosity are the issues of the FLiBe utilization as the coolant. Moreover, the development of precise MHD turbulence model is highly demanded to predict the MHD pressure loss and the heat transfer for fusion reactor design. In this study, the MHD pressure loss and heat transfer characteristics were investigated by means of DNS (Direct Numerical Simulation), and the evaluation of MHD turbulence model proposed by Kenjeres and Hanjalic (2000) and Smolentsev et al. (2002) were also carried out at high Reynolds number (Re) corresponding to the DNS database reported by Satake et al. (2006).

At first, the numerical calculations of MHD turbulent channel flow imposed the wall-normal magnetic field were carried out by using the Kenjeres and Hanjalic (KH) model and the Smolentsev et al. (S) model at the same condition as the DNS data (Satake et al., 2006): Bulk Re = 46000 and Hartmann numbers (Ha) = 32.5 and 65. Compared with the DNS results, both turbulence models can reproduce the MHD pressure loss trend with increase of Ha. However, both models underestimated the turbulent kinetic energy, and the prediction accuracy was getting worse with increase of Ha. Compared with the KH model and the S model, the KH model has a little advantage in the prediction accuracy and this result conforms close to a priori test in MHD source terms in k- and epsilon-equation. To improve the prediction accuracy of the k-e turbulent model, some modification of eddy viscosity would be required. Next, DNS of 2D-fully developed turbulent channel flows imposed wall-normal magnetic field were conducted to investigate heat transfer characteristics. In the computations, thermal properties of the KOH solution (Pr number is 5.2 at 40 deg-C.) were used because the KOH solution instead of FLiBe was used in the JUPITER-2 experiment. Turbulent Reynolds number was kept on 150 and Ha was changed from 0 to 16. The continuity equation, the momentum equations with the electric field described using the electrical potential approach at low magnetic Reynolds number and the energy equation were solved by a hybrid Fourier spectral and the second order central differencing method (Yamamoto et al., 2002). As the results, the velocity profile completely changed from turbulent flow to the laminar one in Ha=13.4. Ha=13.2 was the near critical condition in this Reynolds number of the turbulent-laminar transition. The profiles of the turbulent intensity were in good agreement with the previous DNS results (Noguchi and Kasagi, 1994). As for the temperature fields, the similarity-law between the velocity and the temperature profiles was not satisfied at the near critical Ha condition. This is the reason why the velocity profile was balanced with the Lorentz force term but the temperature profile was balanced with the wall-normal turbulent heat flux at the near critical Ha condition. Present study results suggested that the MHD turbulent model which can consider the anisotropy of the Reynolds stresses and the local change of the turbulent Pr number might be required in the view point of quantitative prediction.

PS3-3018

THERMO-MECHANICAL ANALYSIS OF PEBBLE BEDS IN HELICA MOCK-UP EXPERIMENTS

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In this investigation, a thermo-mechanical model for pebble beds and a method for the identification of the material parameters, recently developed in Forschungszentrum Karlsruhe, are adopted for the analysis of the HELICA (HE-FUS3 Lithium Cassette) mock-up. The mock-up experiments have been carried out in ENEA-Brasimone and the results will be used as a benchmark exercise for the FE codes among the EU associations. A pressure-dependent thermal contact conductance model to represent the pebble-wall interactions has been developed and implemented in FE code. First, the current material model has been verified by uniaxial compression experiments under a wide range of temperature fields, and good agreement between experiments and theory has been achieved. The HELICA mock-up has been modelled by 2-D generalized plane strain elements, and analyzed in ABAQUS. The results show that the temperature and mechanical fields obtained in FE analysis coincide well with the measurements by thermo-couples and LVDTs located at different positions. Furthermore, the effect of a pressure-dependent thermal contact conductance, which can be inferred from the experiments, is included in this FE analysis.

TRITIUM RELEASE FROM LITHIUM SILICATE PEBBLES PRODUCED FROM LITHIUM HYDROXIDE

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Slightly over-stoichiometric lithium orthosilicate pebbles have been selected as one optional breeder material for the European Helium Cooled Pebble Bed (HCPB) blanket. This material has been developed in collaboration of Research Center Karlsruhe with Schott Glass, Mainz. The lithium orthosilicate pebbles are fabricated by a melting and spraying method in a semi-industrial scale facility. In the past, the non-enriched pebbles were produced from a mixture of lithium orthosilicate and silica powders, whereas enriched lithium orthosilicate is not available on the market. Thus, highly enriched lithium carbonate powder was used for the production of the lithium orthosilicate pebbles, which resulted in unsatisfactory pebble characteristics. Enriched lithium hydroxide is commercially available as well, and thus a new production route with lithium hydroxide was pursued. The melting process was found to be applicable to the production of lithium orthosilicate pebbles from lithium hydroxide and silica. The lithium orthosilicate pebbles produced by the process contains oxide phases besides orthosilicate, but it was also found that the oxide phases can be decomposed by annealing at high temperatures. The lithium orthosilicate pebbles produced in this way possesses satisfactory pebble characteristics. Therefore, the authors performed out-of-pile annealing tests using the lithium orthosilicate pebbles irradiated in a research reactor.

The lithium orthosilicate pebbles were irradiated with neutrons in the Kyoto university research reactor. In the out-of-pile annealing experiments, the temperature of the breeder material placed in a tubular reactor made of quartz was raised from ambient temperature to 1173 K at a constant rate of 5 K/min under the stream of various sweep gases. The tritium concentration in the outlet stream of the reactor was traced using two ionization chambers. The ionization chambers were fitted with a water bubbler, which enables to measure the concentrations of molecular form of tritium (HT) and tritiated water vapor (HTO) separately. In the experiments, various sweep gases such as pure nitrogen gas, 1,000 ppm hydrogen/nitrogen gases and 10,000 ppm water vapor/nitrogen gas were used to investigate the effect of the nature of the sweep gas and its composition on the tritium release form the breeder material. The experimental results indicate that almost all tritium was released as tritiated water vapor regardless of difference in the gas composition of the sweep gas. It was also found that a considerably smaller amount of tritium was released when a pure nitrogen gas was used as the sweep gas. Comparison of the experimental results reveals that tritium is released at lower temperatures if the sweep gas contains water vapor.

IRRADIATION TESTS OF A SMALL-SIZED MOTOR WITH RADIATION RESISTANCE

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In the Test Blanket Module (TBM) of the International Thermonuclear Experimental Reactor (ITER), tritium production and release behavior will be studied using neutrons from fusion reactions, as the blanket development for a demonstration (DEMO) reactor. For development of the TBM, in-pile functional tests are planned, including an integrated irradiation experiment of a fusion blanket mock-up for pulsed operation simulating the ITER operation mode, using the Japan Materials Testing Reactor (JMTR) of Japan Atomic Energy Agency (JAEA). Due to be installed in an irradiation rig, a small-sized motor has to be developed for rotating a neutron absorber with a window to realize the simulated pulse operation. Since degradation of materials of the motor may be caused by radiation damage due to neutron and gamma-ray irradiation, it is important to examine the soundness of the motor materials under the neutron and gamma irradiation. In the present study, a small-sized motor with increased radiation resistance was developed as follows. A design of a commercial alternate current (AC) servomotor was adopted in the base structure, and some components of the motor were replaced by those made of radiation-proof materials, through elimination of organic materials. Polyester-coated wire for field coil and epoxy for fixed resin were replaced by polyimide-coated wire and polysiloxane filled with MgO and Al2O3, respectively. Furthermore, inorganic lubricant (Mo-based coating of 4 micro meter in thickness) was treated on the surface of a gear, instead of organic (polyphenylether) oil. Radiation-induced degradation of the components of the developed small-sized motor was examined using JMTR and the Japan Research Reactor No.4 (JRR-4) of JAEA. The motor was operating normally up to a gamma-ray dose of 7x10**8 Gy, a fast neutron (E>1MeV) fluence of 2x10**21 m**-2 and a thermal neutron (E<0.683eV) fluence of 4x10**22 m**-2. The irradiated gamma-ray dose for this motor is about 700 times as high as the operation limit of the commercial motor. Thus, we can conclude that this motor is able to be used for an in-pile test of a fusion blanket mockup for the simulated pulse operation. This motor can also be applied to in-situ tests such as fatigue tests and creep tests in nuclear reactors, as well as to various control mechanisms under radiation environment including neutrons.

VALIDITY OF DISPLACEMENT ENERGY EVALUATION USING MOLECULAR STATICS SIMULATION IN LI2O

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Understanding on radiation damage processes in Li-containing oxides has been regarded as an important subject in fusion blanket engineering, because radiation defects significantly affect the tritium behavior and the material property. The displacement energy is a key parameter that determines the number of defects created by radiation, and thus should be evaluated. However, its determination by experiments has not been done, probably due to difficulties arising from insulating property and complicated crystalline structures of Li-containing oxides.
Molecular simulation is an alternative method to evaluate the displacement energy. Two techniques have been used; one is molecular dynamics simulation (MD) and the other is molecular statics simulation (MS) with the sudden approximation [1]. MD can provide atomic-scale views of radiation events in the dynamics and has been more widely applied. MS seems to provide less reliable results than MD for lack of the dynamics. Nevertheless, its low computational cost could be attractive for application to ternary Li-containing oxides of complicated structures. In the present work, therefore, we aimed to verify how reliable values MS can provide in comparison with MD. Li2O was chosen to be a test material, because Li2O has the simplest structure among Li-containing oxides, which facilitates verification of MS results.
We evaluated threshold displacement energies by MS for a few tens of different irradiation direction, and compared with previous MD results. DL-POLY code was used for MD, while GULP code for MS. In MD, lower threshold energies have been observed for Li than O (20 eV for Li and 50 eV for O on average). This tendency was also realized in MS (15 eV for Li and 40 eV for O), although values were often underestimated by a few tens %.
As for dependence of displacement energy on irradiation direction, MS basically gave results different from MD, not only in quantity but also in quality. It was considered that MS is useful to roughly evaluate a typical value of displacement energy and is not suitable to analyze details, such as its dependence on irradiation direction. Comparison of MS and MD were also done under another potential model, in order to check potential model dependence of the results. No significant dependence was seen.
Some simulations in which MS failed will be analyzed in detailed, and reasons of the failure will be discussed in the presentation.

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X-RAY TOMOGRAPHY INVESTIGATIONS ON PEBBLE BED STRUCTURES

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Granular materials (pebbles) are used in present ceramic breeder blankets both for the ceramic breeder material and beryllium. The thermal-mechanical behaviour of these pebble beds strongly depends on the arrangement of the pebbles in the bed, their contacts and contact surfaces with other pebbles and with walls. The influence of these quantities is most pronounced for beryllium pebble beds because of the large thermal conductivity ratio of beryllium to helium gas atmosphere. At present, the data base for the pebble bed thermal conductivity (k) and heat transfer coefficient (h) is quite limited for compressed beds and significant discrepancies exist in respect to h. The detailed knowledge of the pebble bed topology is, therefore, essential to better understand the heat transfer mechanisms.

In the present work, results from detailed X-ray tomography investigations are reported on pebble topology in i) the pebble bed bulk (which is relevant for k), and ii) the region close to walls with thicknesses of several pebble diameters (relevant for h). At Forschungszentrum Karlsruhe, pebble beds consisting of aluminium spheres with diameters of 2.3 and 5mm, respectively, (simulating the blanket relevant 1mm beryllium pebbles), were uniaxially compressed at different pressure levels. High resolution three-dimensional microtomography (MT) experiments were subsequently performed at the European Synchrotron Radiation Facility, Grenoble.

Radial and axial void fraction distributions were found to be oscillatory next to the walls and non-oscillatory in the bulk. For non-compressed pebble beds, the bulk void fraction is fairly constant; for compressed beds, a gradient exists along the compression axis.

In the bulk, the angular distribution of pebble contacts was found to be fairly constant, indicating that no regular packing structure is induced. In the wall region, the pebble layer touching the wall is composed of zones with hexagonal structures as shown clearly by MT images. This finding is also reflected in the calculated angular contact distribution. With increasing distance from the wall, the regular structure vanishes and the bulk values are approached after the 4th wall layer away from the walls.

Concerning the sum of contact surfaces per pebble, it is shown that the component normal to the compression axis (normal to the heat flow) is approximately independent of the pebble location. For strongly compressed pebble beds, this implies that by extrapolation of the bulk value of k to the wall a further heat resistance might be neglected.

Keywords: Fusion reactor blanket, pebble beds, granular materials, thermal-mechanical behaviour, thermal conductivity, microtomography, void fraction, co-ordination number, contact surfaces

PS3-3028

MEASUREMENT OF TRITIUM PERMEATION IN FLIBE (2LIF-BEF2)

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Interrelated transport processes and chemical interactions characterize the behavior of hydrogen isotopes in molten fluoride salts. In fusion systems hydrogen species in the molten fluoride salts include H2, HT, T2, HF, and TF. Tritium is generated by neutron capture in Li, while hydrogen is present to control the fluorine potential in the gas phase or as a residual product of salt purification. In general, the transport processes are dependent on the following properties: solubility and diffusivity in the molten salt, dissociation at the surface and diffusion in contacting materials, mass transport and possibly recombination at gas-melt interfaces. The chemical behavior is determined to a great extent by the fluorine potential in the molten salt (REDOX condition), which is in turn affected by the radiation environment and nuclear transmutation reactions, by reactivity with container or other contacting materials, by reactivity with the gaseous environment over the salt and by interaction with impurities.
This paper reports on the experimental investigation of tritium permeation in flibe (2LiF-BeF2) at the Safety and Tritium Applied Research facility of the Idaho National Laboratory in the frame of the US-Japan Jupiter-II collaboration. A stainless steel cell formed by two independent volumes separated by a 2mm thick nickel membrane is maintained at temperatures between 500 and 650 degrees Celsius. A controlled amount of T2 gas is flown in excess of argon in the source volume in contact with the bottom side of the nickel membrane, while a thin layer of molten salt is in contact with the top side. The salt has been purified and conditioned to minimize impurities following well-established procedure developed during previous corrosion control experiments, and is maintained in contacts only nickel containers during test and liquid-phase transfer operations. The tritium permeating above the liquid surface is carried by an argon flow to a diagnostic system comprised of a quadrupole mass spectrometer, a gas chromatographer and a proportional counter previously calibrated with D2. Tritium permeability in flibe as a function of temperature is determined by the measured permeation flow rates reached in steady-state conditions, while the diffusivity is determined by fitting the transient process with the analytical solution for the diffusion process. As a result, the solubility of tritium in flibe as a function of temperature is also determined.

FUSION NUCLEAR TECHNOLOGY DEVELOPMENT AT THE PETTEN HIGH FLUX REACTOR

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The High Flux Reactor at Petten is for several decades a major tool in the development of the European Fusion Technology Programme. The HFR is a 45 MW Materials Test Reactor, and operated for about 280 Full Power Days per year.

The important role of the EU in the ITER initiative is also reflected in the HFR programme: HFR's high versatility provides it with extremely relevant R&D capabilities for fusion power plant technology. The HFR contributes to the fusion technology development by providing experimental results utilising the HFR as the neutron source and partners hot cell laboratories to perform post-irradiation testing. The main areas of interest are the ITER vacuum vessel, the blanket development and the development of reduced activation structural materials: chromium steel and ceramic composites. The irradiation of the blanket sections with lithium ceramic pebbles is not limited to post-irradiation testing, but it includes in-pile instrumentation for the operation of the ITER Test Blanket Modules. In this way the HFR provides valuable in-pile process data for blanket operations in ITER.

A wide variety of irradiation projects for structural and functional materials have been undertaken or are in progress.

A major step actually being taken is the transition to testing of components. Examples are irradiation stress relaxation of prestressed bolts, thermal fatigue of primary wall modules, in-pile performance of HCPB pebble-bed configurations and HCLL representative LiPb-Eurofer assemblies. Controlled gas purge with on-line tritium monitoring and triple containment are key features for irradiation of tritium generating specimens. In-pile oxidation of Eurofer has been demonstrated recently. Two high dose irradiation projects concern lithium ceramics and beryllium neutron multipliers key performance issues under DEMO relevant conditions of stress state and neutron fluence levels.

The paper will describe the activities globally and highlight the implications on qualifications of ITER in-vessel parts and the further development of nuclear components for fusion power plants.

TRITIUM TRANSFERS AND MAIN OPERATING PARAMETERS IMPACT FOR DEMO LITHIUM LEAD BREEDING BLANKET (HCLL)

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Within the development of fusion technology, the need of tritium breeding in order to reach fuel self-sufficiency is a major issue. In a fusion reactor, the daily tritium production is of several hundreds of grams and the fluxes containing large amount of tritium need to have a good inventory control.
One difficulty comes from the tritium property to diffuse through hot metallic walls. Because of the double function of the blanket: breeding the necessary Tritium and efficiently extracting the deposited heat, the metallic surfaces used to promote the heat transfer lead also to a non negligible mass transfer of hydrogen isotopes.
In order to improve the management of tritium, different studies have been launched in this field with applications to DEMO breeding blankets and to corresponding ITER TBM.
A general model has been developed to evaluate the potential tritium released by permeation through hot walls up to the secondary circuit with the objective to define the possible ways to reduce the tritium release. The model is applied to the main systems of the Helium Cooled Lithium Lead DEMO blanket.
A system description has pointed out the main circuits and main components involved in mass transfers. For hot walls, Fick laws are written for hydrogen and tritium. This leads to a set of differential equations which are solved in the case of steady state as expected in a DEMO. A sensitivity study is done to determine which the major issues are and where the major reduction can be obtained for minimizing the final tritium release.
This paper describes the used input data and some of them are discussed largely. Among them three main categories have been defined and realistic upper and lower values have been determined from literature and from process analysis. It leads to estimation of tritium release and to determination of the key point where improvement could have a strong impact on the reduction of the Tritium release. At present, permeation barrier are considered, but the chemistry control of Helium coolant could have a stronger impact.
This paper also presents what are the required further improvements of such modelling. A new approach of modelling, coupling system mass balances and finite elements calculations, is presented. It allows taking into account mass flow, convection, temperature exchange in LiPb and in Helium, in particular within the blanket modules, that are essential parameters for achieving a rigorous estimation of the tritium permeation.

MHD SIMULATIONS OF LIQUID METAL FLOW THROUGH A TOROIDALLY-ORIENTED MANIFOLD

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A key feature of liquid metal breeding blankets is the flow manifold, which must distribute/collect the liquid metal coolant among parallel channels. The Dual-Coolant Lead-Lithium (DCLL) blanket concept, as currently envisioned in the US, has several poloidally-running, toroidally-stacked, parallel channels. The blanket requires a manifold section to accept liquid lead-lithium from a central supply, distribute it toroidally, and then feed it into the poloidally-running parallel channels. A similar manifold is required at the end of the parallel channel runs to collect the lead-lithium and merge the flows into a central return line. The design of these manifolds can have a very strong effect on the total pressure drop of the system, as well as on the uniformity of flow distribution between the multiple parallel channels. This later effect, in particular, is of critical concern for the operation of the blanket in order to prevent unacceptable overheating of parallel channels with reduced flow.
In fusion blankets, magnetohydrodynamic (MHD) effects will dominate the pressure drop and velocity profiles of the liquid metal flow in the manifold regions. However, there is very little experimental data available for manifolds with this orientation to the magnetic field – and the data that does exist indicates non-uniformity will increase strongly with increasing magnetic interaction. In order to begin to address these issues for the US DCLL, a series of 3D MHD simulations has been performed at ITER relevant magnetic interaction parameters. The geometry has a single rectangular supply channel, entering a rectangular expansion with field oriented along the expansion direction, finally feeding into 3 rectangular parallel channels stacked in the field direction. These simulations match the range of experimental conditions achievable in a simultaneous experimental test campaign. Various conditions of wall conductivity, geometric variations, flowrate and field strength are explored. Finally, conclusions are drawn about the degree of flow non-uniformity and the best conditions for reducing it to an acceptable level.

PS3-3033

DEVELOPMENT OF HIGH TEMPERATURE LIPB-SIC BLANKET

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A blanket concept based on combination of LiPb, SiC and helium is of particular interests for a demo blanket concept, because it is expected to be feasible in near term target such as TBM, and would eventually achieve high operating temperature above current fission reactors are operated. We propose to use cooling panel made of SiC so that RAFM enclosure is isolated from LiPb breeder and primary heat medium, as designed in DCLL or DFLL designs. The cooling panel made of SiC composite can also work as heat exchanger boundary, and high temperature helium up to 900 degree C is expected as a medium for power generation including possible hydrogen production such as biomass conversion. Staged development strategy can be planned; and RAFM-LiPb blanket with SiC insert such as DCLL or DFLL in TBM can be developed as the first step.

This paper describes recent results of the development of SiC-LiPb blanket in Kyoto University. Fabrication of SiC/SiC cooling panel shows the fine structure for helium channel. High temperature LiPb loop is operational above 600 degree C for experiments on heat transfer, material compatibility, MHD pressure drop, and hydrogen transport. Independent experiments also show hydrogen behavior in this material system, and electrochemical devices are developed to monitor the oxygen and hydrogen concentration in the LiPb. Preliminary design study of the module suggests the feasibility of this concept, and integrated test is planned.

EXPERIMENTAL STUDY OF DECELERATION PROCESS OF TRAVELING WAVE DIRECT ENERGY CONVERTER FOR ADVANCED FUSION

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Advanced fusion is attractive in the view point of utilization of high efficiency direct energy conversion from fusion produced ions. Deutrium-helium-3 reaction is the most possible, however, the energy of created fast proton is so enormous that conventional electro-static converters cannot be applied. Use of a traveling wave direct energy converter (TWDEC), the principle of which was inverse process of a linear accelerator, was proposed for recovering energy of the fast protons. In order to realize the TWDEC, the authors are continuing experimental study by employing a small-scale simulator[1].

A TWDEC consists of a modulator and a decelerator. Fast proton beam extracted from a reactor is introduced in the modulator where radio frequency (RF) electrostatic field modulate the beam velocity, and hence, the protons are bunched and density-modulated in the downstream. The density-modulated protons flow into the decelerator where a number of electrodes connected to a transmission circuit are axially aligned. The flowing protons induce RF current which creates RF traveling voltage on the electrodes. The RF traveling field between aligned electrodes decelerates the protons, thus their energy is recovered into RF power.

In this paper, deceleration process of TWDEC is experimentally examined. In our experimental simulator, because of the small beam current, the induced potential, i.e. the deceleration field is so weak that the beam cannot be decelerated. Thus, we examined the process by dividing into two: one was induction of the deceleration field by the modulated beam, which was called as passive decelerator. The other was energy recovery through interaction between the deceleration field and the modulated beam. In this latter experiment, the deceleration field was supplied externally, and we called this as active decelerator.

As for the active decelerator mode, we performed higher beam energy experiment than previous one. As the beam energy increases, the divergence of the beam is relatively suppressed and efficiency of energy recovery is expected to be improved. In the experiments, the structure of the decelerator was designed and optimized by considering the variation of wavelength of the traveling wave due to beam deceleration. As a result, higher efficiency was obtained compared with the previous lower beam energy experiment. The dependence of the efficiency on the length of the decelerator, however, was not necessarily like those to be expected and showing saturation with some length. This is because the design is based on the ideal deceleration process and the gap between ideal variation of wavelength and experimental one becomes larger in the longer decelerator length.

As for the passive decelerator mode, we have shown the improvement of induction of deceleration field in the previous report[1]. The results on the variation to beam energy and/or modulation will be presented in the symposium.

[1] H. Takeno, et al., Fusion Science and Technology, 43(3) 450-454, 2005.

P3-0001 (Invited)

FUEL CYCLE DESIGN FOR ITER AND ITS EXTRAPOLATION TO DEMO

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ITER is the first fusion device equipped with complete deuterium-tritium fuel cycle that continuously provides fuel to the burning plasma while reprocessing its exhaust to demonstrate the scientific and technological feasibility of fusion power through the pulse plasma operations using 3 kg of tritium on site. All the tritium and deuterium in the exhaust is recovered, purified and returned to the tokamak with minimal delay, so that extended burn can be sustained with limited inventory. Extremely high decontamination factor, that is the ratio of the tritium loss to the processing flow rate, is required for fuel economy and minimized tritium emissions. The ITER main fuel cycle system is composed of fueling & pumping, plasma exhaust processing, isotope separation, and storage & delivery, and is designed to process considerable and unprecedented DT flow rates with high flexibility and reliability. For the safety of the occupational personnel and environment, a multiple barrier concept for confinement with detritiation systems is applied. Due to the nature of the operation and safety, tritium processing systems are required to achieve extremely high performance with high reliability, while complex configuration of the integrated system must be coordinated between the participant teams. Major part of the fusion tritium system will be verified with ITER and its decades of operation experiences. The first part of this paper will report this challenging technology task. Toward the DEMO plant that will actually generate energy and operate its closed fuel cycle, breeding blanket and power train that caries high temperature and pressure media from the fusion device to generation system will be the major addition. Although essential process components will be similar to those to be developed for ITER, tritium inventory control and sustain the fuel supply, and minimizing environmental release will bring another level of technical challenge. Due to the higher burn-up and the difference of the material in the tokamak vessel, a decrease for the main fuel-processing load will be expected. The bred tritium recovery system depends on the blanket concept and material selection of breeder & multiplier, however, at least the dedicated isotope separation system and adequate system for accountancy should be required as a control of tritium production. For the tritium confinement, safety and environmental emission, also blanket, its coolant, and generation systems such as heat exchanger, steam generator and turbine will be the critical systems, because the tritium permeation from the breeder and handling large amount of high temperature, high pressure coolant will be further more difficult than that required for ITER. Detritiation of solid waste such as used blanket and divertor will be another issue for both tritium economy and safety. Fuel and safety issue to be tested and demonstrated in the DEMO will determine the viability of the fusion as a future energy source. The latter part of the paper will review the tritium technology that can be extrapolated from ITER.

PLASMA CONTROL SYSTEMS RELEVANT TO ITER AND FUSION POWER PLANTS

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Since tokamak fusion research has just made a great step forward to an international collaborative project ITER (International Thermonuclear Experimental Reactor), the existing large and medium-size tokamaks are expected to explore more advanced operation scenarios toward the ITER and a future power reactor.
Hence, we believe the following experimental issues are to be adequately investigated, and possibly to be solved in the existing tokamaks: To specify one or more solutions to keep a steady-state plasma with high performance (high beta and high bootstrap current fraction), and to avoid plasma instabilities almost completely. Finally, the developed solutions have to be confirmed using a burning plasma in ITER.
As the above remaining issues are considered the major obstacle to the fusion power plant as common understanding, a plasma control system for ITER should have two important aspects: "Technical inheritance of the currently-working functions" based on the experiences in the current tokamaks, and "flexible or adaptive structure" that could be changed to any new future requirements (the evolution of plasma control systems).
First, we make review on the system configuration and essential functions employed in each plasma control system from the viewpoint of hardware as well as software, at the currently-working world tokamaks (JET, ASDEX-U, Tore Supra, TCV, DIII-D, JT-60, etc.).
Second, we survey ITER control system requirements for the current CODAC design. Conceptual and technical issues are discussed.
Third, flexible structure in the plasma control system should be defined and discussed by assuming the possible future requirements. Technological advance in hardware (computers, I/O devices, etc.) should be considered.
Finally, on the basis of the above discussion, we would like to envisage a future plasma control system for ITER and a fusion power plant.

EVOLUTION OF ITER TRITIUM CONFINEMENT STRATEGY AND ADAPTATION TO CADARACHE SITE CONDITIONS AND FRENCH REGULATORY REQUIREMENTS

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The ITER Nuclear Buildings include the Tokamak, Tritium and Diagnostic Buildings (Tokamak Complex) and the Hot Cell and Low Level Radioactive Waste Buildings. The Tritium Confinement Strategy of the Nuclear Buildings comprises key features of the Atmosphere and Vent Detritiation Systems (ADS/VDS) and the Heating, Ventilation and Air Conditioning (HVAC) Systems. The designs developed during the ITER EDA (Engineering Design Activities) for these systems need to be adapted to the specific conditions of the Cadarache site and modified to conform with the regulatory requirements applicable to Installations Nucléaires de Base (INB) – Basic Nuclear Installations – in France. The highest priority for such adaptation has been identified as the Tritium Confinement of the Tokamak Complex and the progress in development of a robust, coherent design concept compliant with French practice is described in the paper.

The Tokamak Complex HVAC concept for generic conditions was developed for operational cost minimisation under more extreme climatic conditions (primarily temperature) than those valid for Cadarache, and incorporated recirculation of a large fraction of the air flow through the HVAC systems to achieve this objective. Due to the impracticality of precluding the spread of contamination from areas of higher activity to less contaminated areas, this concept has been abandoned in favour of a once-through configuration, which requires a complete redesign, with revised air change rates, module sizes, layout, redundancy provisions and other features.

The ADS/VDS concept developed for the generic design of the ITER Tokamak Complex is undergoing a radical revision in which the system architecture, module sizing and basic process are being optimised for the Cadarache conditions. Investigation is being launched into the implementation of a wet stripper concept to replace the molecular sieve (MS) beds incorporated in the generic design, where concerns have been raised over low reliability due to frequent cycling of large valves (for regeneration of MS beds). R&D to confirm the performance of the stripper concept and to derive comparative data on tritiated water production is being carried out.

In due course a similar overhaul of the tritium confinement concept for the remainder of the Nuclear Buildings (the Hot Cell Complex) will be carried out. This will be initiated once the overall dimensions, layout and other key aspects of the Hot Cell complex have been reviewed in the light of site-specific conditions. The recovery of tritium from high heat flux materials and assay of residual tritium in the waste stream to be transferred to the host for ultimate disposal or processing plays an important role in the tritium confinement strategy for the Hot Cell.

S5-0002

RECENT RESULTS OF R&D ACTIVITIES ON TRITIUM TECHNOLOGIES FOR ITER AND FUSION REACTORS AT TPL OF JAEA

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At TPL (Tritium Process Laboratory) of JAEA, ITER relevant tritium technologies have been carried out. The design studies of Air Detirtiation System have been carried out in JAEA as a contribution of Japan to ITER. For the tritium processing technologies, our efforts have been focused on the R & D of the tritium recovery system of ITER test blanket system. A ceramic proton conductor has been studied as an advanced blanket system. It was observed that the hydrogen transportation rate of the conductor increased with the areas of the triple phase boundary (the contact point of gas, electrolyte and electrode). To decrease the grain size of the metal electrode by sputtering may be a solution for the increase of the areas of the triple phase. A series of fundamental studies on tritium safety technologies not only for ITER but also for fusion DEMO plants has also been carried out at TPL of JAEA. The main R&D activities in this field are the tritium behavior in a confinement & its barrier materials; monitoring; accountancy; detritiation and decontamination etc. The retention of hydrogen on a tungsten surface was studied by exposing a high flux and low energy deuterium plasma. As a new typical observation, a blister bursting with a tail, or a small hole or a vanished cap was found on some grains after the plasma exposure in spite of the low energy plasma. In this paper, the results of above recent activities at TPL of JAEA are summarized from viewpoint of ITER relevant and future fusion DEMO reactors.

S5-0003

DESIGN OF TRITIUM SYSTEMS FOR CHINESE HELIUM COOLED SOLID BREEDING AND DUAL FUNCTIONAL LEAD LITHIUM TEST BLANKET MODULE

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A set of common used ancillary tritium processing systems including the tritium extract system (TES), the coolant purification system (CPS), and the tritium measurement system (TMS) have been designed for the two types of Chinese test blanket modules (TBM) located in a whole test port in the international thermonuclear experimental reactor (ITER), based on the structure features and technical parameters of Chinese helium cooled solid breeder (CH-HCSB) and dual functional lithium lead (DFLL) TBMs. The functions, technical parameters, and process flows of these systems have been described. TES is designed to extract tritium produced in the breeder, to store and separate hydrogen isotope gases. Impurities and tritium in the helium coolant is removed through CPS. TMS has the function of testing the tritium breeding rate in TBM which sometimes replaces the function of TES. Tritium release to the environment from two types of Chinese TBMs and the common tritium systems is well controlled below the tritium safety limit of ITER, based on the tritium permeation and tritium safety analysis.

S5-0004

ADVANCED FUELLING SYSTEM FOR USE AS A BURN CONTROL TOOL IN A BURNING PLASMA DEVICE

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Steady-state Advanced Tokamak (AT) scenarios rely on optimized density and pressure profiles to maximize the bootstrap current fraction. Under this mode of operation, the fuelling system must deposit small amounts of fuel where it is needed, and as often as needed, so as to compensate for fuel losses, but not to adversely alter the established density and pressure profiles. Conventional fuelling methods have not demonstrated successful fuelling of AT-type discharges and may be incapable of deep fuelling long pulse ELM-free discharges in ITER. The capability to deposit fuel at any desired radial location within the tokamak would provide burn control capability through alteration of the density profile. The ability to peak the density profile would ease ignition requirements, while operating ITER with density profiles that are peaked would increase the fusion power output. An advanced fuelling system should also be capable of fuelling well past internal transport barriers. Compact Toroid (CT) fuelling [R. Raman, et al., "Experimental demonstration of tokamak fuelling by compact toroid injection," Nucl. Fusion, 37, 967 (1997)] has the potential to meet these needs, while simultaneously providing a source of toroidal momentum input. A CT is a self-contained toroidal plasmoid with embedded magnetic fields. The 20 Hz injector consists of the formation region, compression, acceleration and transport regions. Fuel gas is puffed into the formation region, and a combination of magnetic field and electric current ionizes this gas and creates a self-contained plasma ring (the "CT"). Then a fast current pulse compresses and accelerates the CT by electromagnetic forces. The accelerated CT will travel at a speed of over 30 cm/μs and for reactors will create a particle inventory perturbation of < 1% per pulse. At this level of particle inventory perturbation, optimized density profiles will not be adversely perturbed. Experimental data needed for the design of a CT fueller for ITER could be obtained on NSTX using an existing CT injector. A conceptual design of a Compact Toroid Fuelling system for ITER will be presented.

S5-0005

SETTING UP AND MANAGING A REMOTE MAINTENANCE OPERATION FOR FUSION

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Trying to set up and manage a remote maintenance operation for a thermonuclear fusion reactor is a complex undertaking.
There are many problems and challenges which need addressing. This paper tries to guide the reader through this process by composing a list of generic problems and by analysing possible solutions. The first challenge before setting up a remote maintenance operation for a fusion reactor is the systematic analysis of all the remote handling requirements. Based upon this the remote handling concept, including facility layout and equipment, can be defined.

The following aspects have to be considered and incorporated into the remote handling concept:

- Remote handling task development
- Remote handling task logistics and resource management

- Command, control and human-machine interface system
- Viewing and camera systems
- Virtual Reality and Augmented Reality software

- Automatic path planning and collision avoidance
- Remote transfer of heavy loads

- Maintainability of RH Equipment
- Reliability, redundant systems and safety
- Rationalisation and modularity in both hardware and software

- Recovery from failure modes
- Condition monitoring & fault detection/prediction
- Ability to deal with unforeseen problems

Oxford Technologies Ltd has a proven track record in setting up and running the Remote Handling group at the JET Joint Undertaking in Culham, UK. Based on the unique experience gained at JET, Oxford Technologies Ltd also developed the current design and remote handling concept of the ITER Hot Cell during a study in 2004. Examples of both the JET Remote Handling experience and the ITER Hot Cell design and layout are given throughout this paper.

S5-0006

DUST CONTROL IN TOKAMAK ENVIRONMENT

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During ITER lifetime, dusts will be produced due to the interaction of the plasma with the Plasma Facing Components. During steady state phase, the dust formation is closely linked to erosion of plasma facing materials and the flaking of the co-deposited layers. The growth of small particles in the edge of fusion plasmas from atomic or molecular precursors which are released by physical or chemical erosion can also lead to dust creation. Particles can as well be created during off-normal plasma events by either condensation and growth of the vaporized material, or pressure-driven ejection of melt layer material or explosive brittle destruction by heating of gas bubbles. At last, during maintenance activities, much larger particles can arise from mechanical abrasion during component replacement. In ITER, these dusts will be activated, tritiated and chemically toxic due to the presence of beryllium. ITER has fixed a set of safety limits in order to control the mobile activation product inventory inside the vacuum vessel, to ensure that the dust chemical reactivity is adequately controlled, and to avoid the hazard of dust explosions. This has been traduced into the following project guidelines: the mobilisable dust inside the vacuum vessel should be limited to 100 kg of W, 100 kg of Be and 200 kg of C (limits based on radiological hazard) and the dusts on the “hot” surfaces of the divertor should not exceed 6 kg of Be, 6 kg of W and 6 kg of C in order to produce less than 2.5 kg of H2 in the vacuum vessel in case of a stream ingress on hot surfaces. Some calculations have shown that the administrative guideline of 100 kg for tungsten dust could be reached in about 500 plasma pulses, and in any case before the assumed replacement of the divertor. Dust diagnostics and removal methods need thus to be developed for ITER considering the constraints linked to magnetic field, radiation, vacuum and temperature. Concerning dust diagnostics, several techniques could be potential candidates in a Tokamak environment. They can be based on erosion, deposition or dust in suspension measurements using optical, sampling or gravimetric systems. The paper will assess the technical feasibility of the different techniques and the adaptation needed, if any, for an implementation in ITER. It will focus in particular on techniques that are being developed to monitor the integrity of the plasma surface components by measuring defects or erosion and that could be applied to dust/deposit monitoring. For the dust removal, three stages are needed: mobilisation of the dusts, collection of the mobilised materials and transport outside the machine. Many studies have been performed in the past years on the collection and transport of the dusts using for example vibrating or electrostatic conveyor. The paper will focus on the studies currently performed in order to improve the mobilisation step using in particular laser beam.

PS4-1003

AN INTEGRATED APPROACH TO THE BACK-END OF THE FUSION MATERIALS CYCLE

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Within the frame of the International Energy Agency (IEA) Co-operative Program on the Environmental, Safety and Economic Aspects of Fusion Power, an international collaborative study on fusion radioactive waste has been initiated to examine the back-end of the fusion materials cycle as an important stage in maximising the environmental benefits of fusion. The study addresses the management procedures for active materials following the change out of replaceable components and decommissioning of fusion facilities.

Numerous differences exist between fission and fusion in terms of activated material type, quantity, activity levels, half-life, radiotoxicity, etc. For fusion, it is important to clearly define the parameters that govern the back-end of the materials cycle. A fusion-specific, unique approach is necessary and needs to be developed. Recycling of materials and clearance (i.e. declassification to non-radioactive material) are the two recommended options for reducing the amount of fusion waste, while disposal as low-level waste (LLW) could be an alternative route for specific materials and components.

Both recycling and clearance criteria have been recently revised by national and international institutions. These revisions and their consequences are examined here with applications to selected studies:

* Recycling: the important radioactive quantities to be limited are contact dose rate, decay heat, and radioactivity concentration. Handling (hands-on, simple shielded, and remote handling approaches), routing related questions (recycling outside the nuclear industry, recycling in nuclear-specific foundries, other possible recycling scenarios without melting), and other issues (C-14, material impurities) are examined.
* Clearance: a definition of a list of nuclides relevant to fusion is made with a proposal of a scenario and a simplified procedure for calculation of a set of fusion-specific clearance limits.
* Disposal: a proposal of a generalized definition of LLW is given, taking into account national and international regulations. As an alternative, less environmentally attractive route to recycling, the LLW disposal in a generic site is considered and its environmental impact is assessed.

FAILURE MODE AND EFFECT ANALYSIS FOR REMOTE HANDLING TRANSFER SYSTEMS OF ITER FE

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A Failure Mode and Effect Analysis (FMEA) at component level was done to study safety relevant implications arising from possible failures in performing Remote Handling (RH) operations.

Autonomous air cushion transporter, pallet, sealed casks and tractor movers needed for port plug mounting/dismantling operation were analysed. For each sub-system, the breakdown of significant components was outlined and, for each component, possible failure modes have been investigated pointing out possible causes, possible actions to prevent the causes, consequences and actions to prevent or mitigate consequences.

Off-normal events which may result in hazardous consequences for the public and the environment have been defined as Postulated Initiating Events (PIEs). Two safety-relevant PIEs have been defined by assessing elementary failures related to the analysed system. Each PIE has been discussed in order to qualitatively identify accident sequences arising from each of them. The two PIEs are:

• RHP Radioactive products (fraction of Dust & T implanted in VV) into Port Cell during RH operations for breach in “VV + cask” isolating boundary.

• RHG Cask stop and radioactive products (fraction of Dust & T implanted in VV) release into Gallery due to Cask leakage during transportation to Hot Cell.

At first glance the consequences of such accidents in terms of radioactive releases should be within the assessment of consequences performed for other studies. Nevertheless, further deterministic analysis could be required to determine response of safety systems (e.g.: efficiency of ventilation systems, isolation of HVAC) and effectiveness of rescue operations in mitigating the consequences and risks for workers. Precisely, even if the two PIEs do not lead to significant radioactive release to the environment, spreading of contamination inside the building and the operating areas can be induced. Consequently, for maintenance and/or decontamination activities, over radiation exposure to workers can be induced.

As an output of this FMEA study, also possible incidental scenarios, where intervention of rescue RH equipments is required to overcome critical situations determine by fault of RH components, were defined and grouped in seven families. Being rescue scenarios of main concern for Remote Handling activities, such families could be helpful in defining the design requirements of port handling systems in general and on transfer cask in particular. Furthermore, they could be useful in defining casks and vehicles to be used for rescue activities.

PS4-1007

CHARACTERISTICS OF HONEYCOMB CATALYSTS TO RECOVER TRITIATED HYDROGEN AND METHANE

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Applicability of honeycomb catalysts to the tritium recovery system was examined considering tritium release accidents in the fusion plant where large volumes of air would be processed by the air cleanup system. Catalytic oxidation of isotopic hydrogen isotopes including tritium is a conventional method for the removal of tritium from air in the working space. However, the high throughput of air causes pressure drop in catalyst beds, which results in high load to the process gas pumping system. The honeycomb catalyst has an advantage in terms of pressure drop, which is estimated to be far less than that in conventional particle-packed catalyst beds. Our previous studies revealed that honeycomb catalysts made of cordierite and Al-Cr-Fe metal alloy substances have preferable oxidizing performance. It was found that the platinum-deposited cordierite catalyst shows the higher oxidation rate for hydrogen gas, and the palladium-deposited metal honeycomb catalyst shows the higher oxidation rate for methane gas. In this study, the properties of honeycomb catalysts were more systematically studied by changing experimental parameters such as noble metal content, mesh density and so forth to obtain design data base for high performance honeycomb catalysts. With regard to catalysts, the amount of noble metal deposited on the honeycomb substrates were varied from 1 g/L to 4 g/L and the mesh density of the honeycombs were changed from 260 to 400 CPSI as well. For operating conditions, the flow rate of the process gases was varied from 0.016 to 0.12 m3/hr, and the concentration of water vapor was changed from 0 to 1.4 %. Results of experimental study suggest that honeycomb catalysts are useful for the treatment of gases with high volumetric velocity in a fusion plant because of their low pressure drop in the catalyst reactor. The platinum catalysts were found to be suitable for oxidation of hydrogen gas, while the palladium catalysts exhibit better performance for oxidation of methane gas. With regard to hydrogen oxidization, the cordierite honeycomb reveals a better oxidizing performance than the metal alloy honeycomb. It was also suggested that the oxidization rate depends on the amount of deposited noble metal but the mesh density does not strongly affects the oxidation rate. Experimental results indicate that the catalytic oxidization rate decreases with increasing moisture content in the experimental gases. It was found that the catalytic activity of palladium catalysts for hydrogen oxidization is substantially decreased under the condition of low temperature and high humidity. In conclusion, it can be said that the honeycomb catalysts are promising alternatives of conventional packed bed type of catalyst for the recovery of tritiated gas if proper noble metal is selected.

DUST EXPLOSION HAZARD IN ITER: LIMITING OXYGEN CONCENTRATION MEASUREMENTS OF ITER-RELEVANT DUSTS

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The work aim is to support inert-gas dilution mitigation methods proposed recently to prevent accidental explosions of dusts accumulated inside the ITER vacuum vessel during the machine operation. A standard method of 20-l-sphere is used to test 4 micron graphite dust and 1 micron tungsten dust to measure their Limiting Oxygen Concentration (LOC) values, below which no self-sustained flame propagation is possible in the dust cloud at any dust concentrations. The tested dust clouds are formed inside the 20-l spherical explosion bomb in nitrogen-diluted air atmospheres at normal initial conditions. The oxygen content in the pre-explosion atmospheres is varied from normal (about 21 vol. %) down to 9 vol. %. The tested dust cloud concentrations are 150 - 300 g/m3 of the graphite dust and 3000 g/m3 of the tungsten dust. The dust clouds are ignited with 2, 5, or 10 kJ igniters. The dependences of maximum overpressures and maximum rates of pressure rise generated in course of the dust cloud explosions are measured as functions of oxygen content in the pre-explosion atmospheres. The maximum overpressure generated by the graphite dust clouds ignited with 2 kJ reduces from 4 bar at normal oxygen content to 0.5 bar at 17 vol. % O2. In case of a stronger 10 kJ ignition the maximum overpressure is higher 0.5 bar down to 11 vol. % oxygen.

The tungsten dust ignited by 5 kJ generate 4 bar overpressure in normal air. With reducing oxygen content the overpressure decreases to 0.5 bar at 13 vol. % oxygen. However, the observed regimes of the tungsten dust explosions seem to be overdriven under the tested conditions because of too high ignition energy (5 kJ) for rather a small combustion volume (20 l). Even the value of 15 vol. % oxygen can be considered as a conservative estimate of LOC for 1 micron tungsten dust.

The results obtained indicate that only a weak dilution of accidental atmosphere in ITER vacuum vessel can suppress the dust explosions in case of severe accident.

Keywords: ITER safety, dust explosion

DUST RESUSPENSION AND TRANSPORT MODELING FOR LOSS OF VACUUM ACCIDENTS

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Plasma surface interactions in tokamaks are known to create significant quantities of dust, which settles onto surfaces and accumulates in the vacuum vessel. In ITER, a loss of vacuum accident may result in the release of dust which will be radioactive and/or toxic, and provides increased surface area for chemical reactions or dust explosion.
A new method of analysis has been developed for modeling dust resuspension and transport in loss of vacuum accidents. The aerosol dynamic equation is solved via the user defined scalar (UDS) capability in the commercial CFD code Fluent. Fluent solves up to 50 generic transport equations for user defined scalars, and allows customization of terms in these equations through user defined functions (UDF). This allows calculation of diffusion coefficients based on local flow properties, inclusion of body forces such as gravity and thermophoresis in the convection term, and user defined source terms. The code accurately reproduces analytical solutions for aerosol deposition in simple laminar flows with diffusion and gravitational settling. Models for dust resuspension are evaluated, and code results are compared to available resuspension data, including data from the Toroidal Dust Mobilization Experiment (TDMX) at the Idaho National Laboratory. Extension to polydisperse aerosols and inclusion of coagulation effects is also discussed.

D-T NEUTRON STREAMING EXPERIMENT SIMULATING NARROW GAPS IN ITER EQUATORIAL PORT

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There are gaps between the vacuum vessel port walls and the port plugs in ITER, providing possible radiation streaming paths. The dimensions of the gaps are ~2 cm in width and ~200 cm in depth and offset geometries are arranged on the middle of gaps to mitigate the streaming effect. The neutron streaming through gaps have been evaluated in the ITER machine design with analytic tools, such as Monte Carlo code in order to assure low enough radiation dose rates after shutdown for allowing personnel access. In the past, experimental verifications were made, simulating gaps among the blanket modules with about 50 cm depth, but not with full depth to the outside of the vacuum vessel. Therefore, further experimental verification with the full depth of the gaps is necessary. Neutronics experiments with a thin slit assembly simulating the gap between the port wall and the port plug have been conducted using the Fusion Neutronics Source (FNS) of the Japan Atomic Energy Agency under the ITER/ITA Task 73-10.

The experimental assembly with a slit of 2 cm in width, 195 cm in depth and 3-cm offset at 56-cm depth from the surface was constructed with iron blocks. The distance from the D-T neutron source to the surface of the assembly was 20 cm. D-T neutrons were generated by 350-keV and 1-mA deuteron beam and tritiated titanium target (0.37 TBq). The intensity was about 10 to the power 11 neutrons/s. In order to evaluate distributions of neutron fluxes along the slit as a function of the depth from the assembly surface, fission rates were measured by U-238 and U-235 micro-fission chambers. The experimental errors of these fission rates were within 10%.

This experimental result was analyzed by using the Monte Carlo code MCNP-4C and 3-dimensional Sn code ATTILA with the nuclear data library FENDL-2.1. JENDL-Dosimetry file 91 was adopted as U-238 and U-235 fission cross section data.

From our experiment and analysis, the following facts were found: (1) Experimental data of U-238 and U-235 fission rates were obtained up to 116 cm in depth successfully; (2) The calculation results with MCNP-4C agreed well with the measured ones; (3) Preliminary calculation results with ATTILA showed considerable underestimation at the backward region through the offset structure.

BIOLOGICAL HAZARD ISSUES FROM POTENTIAL RELEASES OF TRITIATED DUST FROM ITER

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Tritiated dust will be generated during the operation of ITER. Several dedicated studies, having performed in-vitro and in-vivo experiments on metal tritide and carbon tritide dust, raised some concerns about the protection guidelines for workers exposed to tritiated dust, if based on the radiotoxicity of tritium in form of tritium gas (HT), tritiated water (HTO) or organically bound tritium (OBT). While the behaviour of HT, HTO and OBT in the human body is well understood, the same is not fully true for tritiated dust, considering the different size distributions and the variety of base materials (through density and morphology), just to cite the most important ones. Inhaled HTO is translocated to blood completely and instantaneously, and then distributes uniformly throughout the body without changing its chemical form. About 1% of the inhaled HT is dissolved in body fluids and tissues and out of this fraction about 1% is then converted to HTO and the rest exhaled. The in-vivo and in-vitro studies on tritiated dust have shown the dependence of the tritium clearance and retention in the human body from their physico-chemical parameters. The most important ones are:

• The dust particles size, influencing the dust deposition pattern and the self absorption fraction of beta rays which in turn affects the dose delivered to lungs;
• The particle density influencing the tritium dissolution fraction and the self absorption fraction of betas particles;
• The dust particle specific surface area influencing the tritium dissolution fraction.

Therefore, it is important to develop credible and sound biokinetic models and from there dose conversion factors for tritiated dust for materials considered in ITER.

From studies carried out so far, it is evident that absorption to lung differs from other tritium forms. Tritiated dust ranges from absorption type S (slow) to type M (moderate) according to the ICRP classification, whereas HTO and HT are classified as F (fast). Some recent studies on JET dust relative to graphite and CFC dust have dealt with physico-chemical characterization and with in vitro tritium dissolution studies. Some uncertainties still remain and further testing is necessary.

In the working areas of ITER the radioactive dust concentration in air will be kept at a very low level. However in an incidental sequence, the atmosphere inside the buildings, including operative areas as well, might get contaminated with small amounts of tritiated dust and hence inhalation risks cannot be excluded.

The present paper summarizes the results from previous investigations on the subject which can build the basis for deriving sound dose conversion factors for tritiated dust aimed for internal dosimetry. At the same time, it proposes further studies and experiments in order to have a complete understanding of the biological hazards of tritiated dust for ITER relevant materials.

EXPERIMENTAL AND DESIGN ACTIVITIES ON WDS AND ISS AS EU CONTRIBUTION TO ITER FUE

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The Water Detritiation System (WDS) of ITER is one of the key systems to control the tritium content in the effluents streams, to recover as much tritium as possible and consequently to minimize the impact on the environment. In order to mitigate the concern over tritium releases into the environment during pulsed operation of the Torus, the WDS and Isotope Separation System (ISS) will be operated in such way that WDS will be a final barrier for the processed protium waste gas stream discharged from ISS. The ITER ISS consists of a cascade of four cryogenic distillation columns with the aim to process mainly two gas streams, one from Torus Exhaust Processing (TEP) and other from WDS mixed with the returned stream from Neutral Beam Injectors (NBI). The behaviour of the CD cascade has to be characterized with high accuracy in view of thermal and isotopic fluctuations during Torus pulses.
To support the research activities needed to characterize the performances of various components for WDS and ISS processes in various working conditions and configurations as needed for ITER detailed design, an experimental facility called TRENTA based on the combination Combined Electrolysis Catalytic Exchange (CECE) – Cryogenic Distillation (CD), representative of the ITER WDS and ISS protium separation column is in operation at TLK.
The CECE process consists of a solid polymer electrolyser unit as envisaged to be used in ITER WDS, and an 8 m Liquid Phase Catalytic Exchange Column (LPCE). The CD system consists of a refrigeration unit of 250 W cooling capacity at 16 K and a cryogenic distillation column of 55 mm in diameter and 2.7 m as process lengths.
The experimental program on the TRENTA facility is focused on two major issues:
- To investigate the separation performances and liquid hold up of different packings potentially to be used on cryogenic distillation process and the separation performances of the catalyst/packing for the LPCE process.
- Trade off studies between the CECE and CD processes during isotopic and thermal transitory regimes. The experimental data obtained on TRENTA facility will allow improving the dynamic modelling code TRIMO and benchmark against ITER relevant operation conditions.

The preliminary experimental results concerning the separation performances of the 8 m LPCE column and the hold up of the reference packing for ITER ISS will be presented.
An evaluation of the impact on the design and operation of ITER WDS and ISS will be provided as well.

EXPERIMENTAL STUDY OF THE ITER VDS CATALYST POISONING

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The catalyst of the ITER Vent Detritiation System (VDS) has to oxidise the tritiated gases and trap the resulting tritiated water, in order to continue to provide the tritium confinement function of the VDS: an experimental study has been carried out in order to determine its ability of to operate when exposed to the products of combustion released during a fire.
In our tests the VDS catalytic recombiner has been tested in presence of fumes generated by the combustion of selected materials (polyvinyl chloride, methyl methacrylate, vacuum pump oil and polytetrafluoroethylene). These materials have been burnt in an oven at 200 °C: the arising combustion fumes have been sent into a catalytic bed where a Pt 0.5% on alumina catalyst operated the conversion of the tritiated gases into tritiated water at 400 °C with a spatial velocity of 6000 h-1. The catalyst efficiency has been assessed by adding tritium as a tracer to the combustion fumes entering the catalytic recombiner and measuring the activity of the tritiated water collected after this reactor.
The studied catalyst has been mainly affected by the fumes coming from the combustion of polyvinyl chloride: the measured catalyst efficiency has been 86.7 %. Especially, in this case the presence of chlorides would have impaired the detritiation process by poisoning the catalyst.
The combustion tests with methyl methacrylate and vacuum pump oil with polytetrafluoroethylene have shown slither reduced catalyst efficiencies of 91.1 and 93.5, respectively.

Keywords: VDS catalyst, catalyst poisoning, combustion tests.

THE DEVELOPMENT OF THE STANDARD OPERATING PROCEDURE FOR THE SDS IN TRITUM PLANT

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The SDS (Storage and Delivery System) is one of the systems consisting of the tritium fuel cycle in the ITER tritium plant. The main purpose of the SDS is to store the deuterium and tritium and to supply the gases needed for operation of the ITER. In this paper, the standard operating procedure of the SDS is developed in satisfying the functional requirement. The standard operating procedure of the SDS consists of 9 operating modes, which are (1) fuel initial loading, (2) fuel supply during plasma operation, (3) fuel circulation during plasma operation, (4) recovery of fuel (T2, DT, D2) from ISS, (5) in-bed calorimetric measurement helium loop, (6) helium-3 recovery, (7) over pressure protection, (8) ZrCo bed regeneration, and (9) vacuum service line arragement in the SDS. For each operating mode, the definition, the preparation and prerequisites, and the operating procedure are described. According to this standard operating procedure, the operating flow diagrams are drawn newly and the P&ID based on the FDR-2001 is revised.

EVALUATION OF THE IMPACT OF ABLATION LOSES IN THE PELLET INJECTOR SYSTEM OF ITER ON THE ISOTOPE SEPARATION SYSTEM

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In view of several design change proposals as a result of R&D activities on different systems of the ITER Tritium Plant and systems which interface with Tritium Plant and confinement systems, an up-date of these interfaces appears to be rather critical. This activity was started inside the EU participating team and several issues with impact onto the design of the fuel cycle systems have been identified.
The experience with pellet injector systems on several fusion devices, i.e. ASDEX-upgrade, have shown that the absolute pellet delivery efficiency, namely number of pellets delivered to the plasma versus launched pellets, is approximately 80 %. In addition to this pellet loss a mass loss in the flight tube due to pellet ablation needs to be considered to estimate the overall fuelling efficiency. Since only a part of the initial pellet particles are injected into the plasma there is an inconsistency between amount of tritium delivered from the tritium plant and that required from plasma. However, the way to recover ablated gas is not yet clear in details.
The ablated and recovered gas during operation of the pellet injectors may require reprocessing within the tritium plant to get the required composition of the gases used for fuelling. The amount of gas ablated and recoverable within the pellet injectors itself will also depend on the design option of the pellet injector.
ITER is now considering a design change of pellet accelerator from a centrifuge to a pneumatic system. In the case of a pneumatic accelerator the composition of the propellant gas, initially pure deuterium, in the buffer vessel of the propellant system will change with time due to the mixing with ablated tritium or DT and the option of sending the gas processed in the pellet injector back to the tritium plant has to be considered.
In the 2001 ISS configuration there was no dedicated stream from the fuelling system to be processed within the ISS. A modeling of the ISS was performed with the aim to quantify the impact of the additional stream from pellet injector system on the tritium inventory in the cryogenic distillation cascade. The paper will present the increase in the ISS tritium inventory for eight fuelling scenarios considering both the cases when 50%DT or 90%T2 will be used as gas for the pellet injector. The influence of this additional stream to the ISS functioning, time necessary for processing will also be discussed.

TOKAMAK EXHAUST PROCESS FOR THE ITER PROJECT

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The ITER project calls for an unprecedented amount of hydrogen isotopes to be processed. To facilitate environmental responsibility and economic application of fusion technology, the re-use of hydrogen isotopes is vital. The US ITER Project Office (USIPO) is responsible for the front end of the ITER Tritium Plant, the Tokamak Exhaust Processing (TEP) System. The TEP system must separate the Tokamak exhaust gases into a stream containing only hydrogen isotopes and a stream containing only non-hydrogen gases.

The USIPO has selected the Savannah River National Laboratory (SRNL) in partnership with the Los Alamos National Laboratory (LANL) to complete the TEP portion of the project. SRNL's participation builds on the laboratory's decades of work with hydrogen and its isotopes deuterium and tritium - providing the applied research and development that supports the Savannah River Site's handling of tritium. SRNL's experience and expertise in large-scale tritium processing systems and its track record of effective project execution are a unique combination that is key to the success of the ITER project. LANL brings to the partnership experience and expertise in tritium processing technologies specific to the fusion program. This knowledge and understanding were gained through the development and operation of the Tritium Systems Test Assembly at Los Alamos for over 20 years starting in the late 1970's.

The US's implementation of the tokamak exhaust processing (TEP) system will provide a technically mature, robust, and cost-effective solution for the separation of hydrogen isotopes from the tokamak exhaust stream. The TEP technology, design challenges, and project status will be presented.

HYDROGEN ISOTOPE SEPARATION CAPABILITY OF MORDENITE COLUMN FOR GAS CHROMATOGRAPH

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In a nuclear fusion reactor system, a monitoring of hydrogen isotopes including tritium is necessary from the viewpoint of system safety control. So, the development of the methods for hydrogen isotope measurement is important issue. A gas chromatography using a cryogenic separation column is one of the methods for hydrogen isotope analysis. However, use of a refrigerant such as liquid nitrogen is a cause of long analysis time and is not suitable for easy installation. The development of the column material having separation capability at fairly high temperature region (about 200K) is one of the solutions for these weak points, because this temperature region can be reached by an electrical device. Synthesis zeolite is a probable candidate of the separation column material. Its structure varies by the ratio of silica to alumina, the kinds of cation and so on, and it gives the unique function to each zeolite. If the factor effected to the hydrogen adsorption characteristics of the synthesis zeolite is clarified, it may lead to the development of the new zeolite optimized to the separation column. So, investigation of hydrogen adsorption property of various synthesis zeolite is necessary. Mordenite (MOR) is a kind of the synthesis zeolite, and it has been reported that the separation column using MOR has possibility to separate hydrogen isotope mixture at fairly high temperature. However, hydrogen adsorption property of MOR is not so clear. Therefore, the present author has investigated the hydrogen adsorption capacity of Mordenite (MOR) at various temperatures, and has proposed their adsorption isotherms. In this work, the separation columns using MOR were made and tested. The column size was 3.0mm in the inner diameter and 1600mm in the length, and the particle size of MOR packed into the column was adjusted between 80 and 100 mesh. Neon was used as the GC carrier gas, and its flow rate was adjusted to 50cm3/min (STP). When the sample gas of 5cm3 including 90ppm of H2 and 170ppm of D2 at 1atm was introduced to this column, the peaks of H2 and D2 were mostly separated at 175K, but they were not separated at 194K. MOR column adjusted in this work was still not for the practical use. However, this result suggests the possibility of the existence of the synthesis zeolite which can separate hydrogen isotope mixture at fairly high temperature. Further investigation is necessary.

THERMAL RELEASE OF TRITIUM FROM SS316

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In an effort to improve current understanding of the mechanisms controlling the long-term release of tritium incorporated thermally into stainless steel SS316 and to develop reliable as well as economically feasible techniques for the conditioning of tritium-containing metallic wastes, a systematic investigation is underway in Toyama under carefully controlled conditions.

The release rate of tritium from SS316 at ambient pressure was determined experimentally in a flow system at several constant temperatures within the range 287 ? 573 K for rather extended periods of time. Under these conditions HTO was found to constitute by far the most important tritium-containing species being released, i.e. approx. 99 %. Much data has accumulated in recent years with a variety of specimens, i.e. type of stainless steel and specimen dimension, loaded with tritium under different pressure and temperature conditions.

Dynamic behavior of long-term tritium release has been successfully modeled using a one-dimensional diffusion equation and assuming that the release rate is governed by the tritium flux at the metal surface boundary. The implications of the results for interim storage and thermal conditioning of stainless steel waste will be discussed.

TRITIUM RELEASE FROM BERYLLIUM MATERIALS UNDER THE REAL OPERATION CONDITIONS

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Beryllium will be used both as neutron multiplier and as plasma facing material in future fusion reactors and will be subjected to action of radiation, temperature, electric and magnetic fields simultaneously. Tritium is generated in beryllium by its reactions with neutrons and by action of plasma under operation conditions of the reactor and will accumulate in both beryllium materials - pebbles and tiles. The tritium retention may cause fuel problems and environmental hazard. Tritium release at thermo-annealing of the beryllium pebbles irradiated in the BERYLLIUM experiment, and the beryllium tiles from JET (exposed to D+D, D+T plasma) was investigated under 5 MeV fast electron radiation of the dose rate 14 MGy/h and in magnetic field of 1.7 T separately and simultaneously in order to evaluate possible effects of these factors. Also, tritium sorption and desorption of unused beryllium tiles were investigated to evaluate action of radiation and magnetic field. Chemical forms of tritium and their distribution were determined in the beryllium samples with lyomethods. Chemical forms of tritium in the beryllium pebbles and tiles both not treated and treated are similar - T2, T+, T°, but their abundances and their distribution are different in the volume of the samples. Tritium release from the irradiated beryllium pebbles and tiles decreased slightly and increased, respectively, in the presence of magnetic field. Irradiation with fast electrons facilitated the process of tritium release, but all three factors together considerably increased the tritium release at thermo-annealing. Irradiation with fast electrons stimulated thermo-sorption of tritium about 0.1 Pa at 773 K for 3 h in unused beryllium tiles, but the simultaneous action of magnetic field and radiation did not change sorption. The simultaneous action of radiation and magnetic field increased the tritium desorption at 773 K for 0.5 h by a factor of 6.

IMPROVED CHARACTHERISTICS OF HYDROPHOBIC POLYTETRAFLUOROETHYLENE-PLATINUM CATALYSTS FOR TRITIUM SEPARATION

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This paper presents recent results concerning the new preparation method and charateristics of the hydrophobic catalysts used in hydrogen isotopes separation. The objectives of the paper are: (1) to assess the curent status and find a new procedure for the preparation of a new improved hydrophobic catalyst; (2) to improve the characteristics and performances of platinum hydrophobic catalysts.

As result of review of references it was concluded that platinum is one of the most active and efficient catalytic metal and polytetrafluoroethylene it's a very good wet-proofing agent in the hydrogen isotopes separation processes.

A category of new improved hydrophobic Pt-catalysts has been proposed, prepared and characterized and are now under testing. The novelty consists in adding of metallic oxides as a new binding and wet-proofing agents (titanium dioxide, zirconium dioxide). The added metallic oxides play a catalytic role, too. The physico-structural parameters of the improved catalyst have been determined and are discussed. Preliminary tests concerning the catalytic activity and stability in tritium separation process have been carried-out. The new proposal is a promising idea to improve the performance of conventional hydrophobic Pt-catalysts.

BEND POINTS OF HYDROGEN PARTIAL PRESSURE CURVES OBTAINED BY TRITIUM REMOVAL SIMULATION TESTS

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In previous studies, a conceptual tritium cleanup system was developed to remove tritium in various chemical forms from exhaust gases. This cleanup system differs from conventional ones employed at many facilities where tritium is handled in that tritium is removed as tritiated hydrogen molecules. In conventional systems, the tritium is removed in the form of water vapor. The present system consists of five main components: hydrogen separator, decomposition-processing vessel, hydrogen-absorbing vessel, circular pump, and buffer tank. The decomposition-processing vessel of the five main components was developed for this system in our laboratory.

For the present study, it was assumed that the gas for processing was composed of hydrogen, methane, and helium. Some of the hydrogen and methane molecules were assumed to be tritiated, and helium was added to maintain a steady flow of gas through the stages of processing in the cleanup system. The performance of the tritium cleanup system was examined by computer-based simulation with simulated exhaust gas. To monitor removal of tritium from the exhaust gas, the partial pressures of hydrogen and methane were examined in a series of performance tests by computer simulation.

Results indicate that the partial pressure of hydrogen underwent a characteristic change with two bend points, even though the partial pressure of methane continued to decrease steadily. The time positions of the bends and time interval between the two bend points on the curve of hydrogen partial pressure were investigated under various conditions in relation to component fraction and gas volume.

The first bend appears to correspond to the completion of one cycle of processing and the second bend may correspond to the period where helium becomes the dominant determinant of pump speed. Both bends were observed at longer times, and the time interval between both bends increased as the amount of gas to be processed increased. Similarly, the second bend was observed at longer times, and the time interval between both bends increased with methane percentage. The results suggest that some features of exhaust gas can be estimated by inspecting both bend points, because the positions of the bend points and the time interval between them provided information about exhaust gas components and volume.

STUDY ON THE TECHNOLOGY OF CECE-GC SYSTEM FOR WATER

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Based on the researches progress of the overall project, electrolysis concentration, gas-liquid catalytic exchange, hydrogen-oxygen combination, gas chromatography for hydrogen isotope separation, tritium storage and so on. A CECE-GC experimental system, which system's disposal capability of 10 ton/a of tritiated heavy water was set up in the middle of 2004 at China Academy of Engineering Physics (CAEP). It has been fulfilled demonstration experiment of recovering deuterium from water contended deuterium and simulation operation of recovering tritium from tritiated heavy water with tritiated water. The system has been operated for 240h,the total concentration factor of CECE is about 4,and the separation factor of tritium is around 10. The GC system of 50m3/d has been recovered 90% hydrogen, which tritium concentration was depleted more than 1000,from 10m3 tritiated hydrogen in 6h.This experimental system is a important basis for further engineering research.
Key words: Heavy water, tritium Extraction, CECE-GC, Demonstration experiment

ENGINEERING DESIGN AND R&D OF IMPURITY INFLUX MONITOR (DIVERTOR) FOR ITER

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The main function of the Impurity Influx Monitor (divertor) is to measure the parameters of impurities and hydrogen isotopes (tritium, deuterium and hydrogen) in the divertor plasma for controlling the plasma by using spectroscopic techniques in the wavelength range of 200 - 1000 nm which is used in present tokamak experiments because no vacuum extension is necessary. The expected impurities are carbon, tungsten, beryllium and copper originating from the divertor target plate and from the surface of the first wall in the main chamber. In ITER, the measurements are required for the full duration of the ITER pulse (> 600 s) and special provisions are necessary to measure in the harsh environment for diagnostic components such as high temperature, high magnetic field, high vacuum condition and high radiation field.

The optical design of Impurity Influx Monitor (divertor) was carried out for the new ITER design. To simplify the optics, a simple Cassegrain telescope composed of simple spherical mirrors and lenses is used as the collection optics. In addition, a micro lens array will be inserted just in front of the fiber bundle to expand the observed area toroidally to increase the light detected. Ray-tracing analysis shows that the spatial resolution of ITER requirement (50 mm) will be achieved. The designed optics is integrated in the port plug. As a result, front-end optics in the upper port can be installed inside the pipe of inner diameter of 300 mm and the optical components in the equatorial port are arranged to avoid many other diagnostic equipments. Detailed mechanical design of front-end optics was also carried out. In the port plug, it is necessary to reduce the temperature rise caused by the nuclear heating for optical components. Heat analysis was carried out for optimization of the cooling method of mirrors, mirror holders and mount modules. As a result, mirrors can be cooled by the thermal conduction keeping sufficient heat flow by the thermal conductivity and making the cooling channels on the mount module uniformly. The effect of thermal strain to the optical properties was also calculated by using the optical design code. In the optics on the upper port, it is small. But in the equatorial port, the image on the imaginary plane is displaced about 150 mm by the difference of the thermal strain. From this result, monolithic construction such as the front-end optics of the upper port is favorable in the point of view of the cooling.

R&D of optical components and optical systems used in this system is ongoing now. For example, the prototypes of metal mirrors made of molybdenum and aluminum, micro lens array and a Cassegrain telescope with the alignment optics have been produced and the optical test will be carried out. Results will be shown in this conference.

EXPERIMENTAL DURABILITY STUDIES OF ELECTROLYSIS CELL MATERIALS FOR WATER DETRITIATION SYSTEM

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The radiation durability of the solid-polymer-electrolyte (SPE) water electrolyzer composed of the Water Detritiation System (WDS) was investigated. A series of gamma-ray and electron beam irradiation tests of Nafion N117 ion exchange membrane, a key polymer in a SPE electrolyzer, beyond ITER-WDS requirement (530kGy) indicated Nafion N117 has enough radiation durability up to 1600 kGy from the view points of mechanical strength and ion exchange capacity. A gamma-ray irradiation test of the whole commercial SPE cell up to 530kGy indicated entire loss of electrolysis function mainly because of the degradation of PTFE for insulator. To keep electrolysis function of a SPE cell up to 530 kGy, we suggest replacing PTFE with polyimide. A gamma-ray irradiation test of soaking Kapton polyimide observed no serious damage in strength up to 850 kGy. Concerning rubber material for O-ring seal, we observed that soaking VITON rubber keeps the constant value of tensile strength up to 850 kGy. Moreover organic elution was not observed from a soak of VITON. From the viewpoint of stable strength and organic elution, VITON is a first candidate for rubber material.

PERFORMANCE OF ELECTROCHEMICAL HYDROGEN PUMP OF A PROTON-CONDUCTING OXIDE FOR THE TRITIUM MONITOR

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The National Institute for Fusion Science (NIFS) has a plan of deuterium plasma discharge using Large Helical Device (LHD) in the near future. Under the deuterium experiment conditions, it is estimated that 430 MBq of tritium will be generated in each discharge shot, involving the injection of a neutral deuterium beam into deuterium plasma. The exhaust gas containing tritium will be disposed by the tritium recovery system and the disposal gas will be released from the stack. From the viewpoint of the safety management and the public acceptance, we have to monitor the low level of tritium in the exhaust gas and working environment. Therefore, we have proposed new tritium monitor by means of recovery of hydrogen isotopes and removal of natural radioactive isotope such as the radon gas in air.1)
As a candidate material for hydrogen isotopes recovery, we have examined the application of proton-conducting oxides which have a function of electrochemical hydrogen pump at elevated temperature. They have attractive advantages such as: hydrogen extraction from hydrogen molecules and hydrogen compounds; control by electric current; no pressurization of supplied gas: hydrogen pump from low concentration gas to high concentration; strong against irradiation compared to organic polymer electrolytes; tritium is treated in the form of gaseous hydrogen.2)
For feasibility study, we manufactured an apparatus of the prototype tritium enrichment system for the tritium monitor equipped with a proton-conducting oxide and a closed loop system. Then, we carried out the performance tests of the prototype one-end-closed tube made of CaZr0.9In0.1O3-a, which is superior to chemical stability and mechanical strength. The shape of the test tube was 14 mm in outer diameter, 12 mm in inner diameter and 340 mm in length. The platinum electrode was attached on both sides of the test tube and the effective area on the cathode electrode was 68 cm2. In this experiment, wet argon gas containing with 1.2 % water vapor was fed to the anode and dry argon gas (water vapor < 0.02%) was fed to the cathode at 100 ml/min, respectively. The sample was heated up to 973K by an electric furnace. Then, the constant voltage was applied between the electrodes by a potentiostat. As the result, the maximum hydrogen evolution rate and the current were 0.95 ml/min and 0.22 A at 4.5 V. The recovery rate of hydrogen was 80%. And we also evaluated the performance of hydrogen concentration by a closed loop system of the prototype tritium enrichment system. Hydrogen gas could be concentrated until 4.1 % in 100 min at 4.0 V, 973 K. The latest development status of a prototype proton-conducting oxide for the tritium monitoring system will be presented.

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OPTIMISATION OF NEAR-TERM PPCS POWER PLANT DESIGNS FROM THE MATERIAL MANAGEMENT STANCE

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The effective management of active material arising from fusion power generation is of crucial importance to maximise the environmental benefits of fusion. In recent years, several EU and international activities have focused towards minimising fusion waste and its radiotoxicity. Reviews have been made of industry practices and international standards to support a comprehensive management strategy based on maximum clearance, recycling and refurbishment of materials. Following this effort, the next step is to optimise the power plant designs according to this strategy and following the “low-activation-design” philosophy of earlier studies.

In this paper, the design of two near-term PPCS plant models based on ITER-relevant technology, a helium-cooled pebble bed and lithium-lead blanket concepts, are re-visited to optimise the management of active materials and minimise wastes. Combined use of novel shielding materials, customised radial builds and impurity control achieve maximum clearance and recycling potential of the irradiated material, and minimise the radiotoxicity of any residual secondary wastes. Up to 17% of the material can achieve clearance before 100 years, representing the majority of the decommissioning stream. Of the remaining material, most can be recycled in conventional nuclear foundries. C-14 generation can be reduced by at least 95% with adequate control of nitrogen impurities. Results confirm the trends obtained in previous work pointing to over-conservatism of the original PPCS analyses based on out-of-date criteria and experience.

PARAMETRIC ASSESSMENTS ON HYDROGENIC SPECIES TRANSPORT IN CVD-DIAMOND VACUUM WINDOWS USED IN ITER ECRH

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Insulators used as H&CD and Diagnostic vacuum windows (VW) in ITER may become modified by surface intake and bulk transport of hydrogenic species. VW, operating under severe radiation levels, have a primary safety role as tritium confinement barriers. Ionizing radiation enhances the (H') uptake and release at surfaces and diffusion rates in the bulk. Radiation damage modifies the material´s bulk trapped inventories by increasing steady state trapping centre concentrations. An experimental programme is ongoing at CIEMAT, to quantify radiation effects on H transport characteristics and also the possible impact on the VW. The reference material for ECRH VW is CVD diamond.

As a parallel activity, parametric transport assessments are being made in order to obtain a wide evaluation of permeation fluxes, ranges, and soluted/trapped inventories in CVD diamond. Transport models have been developed based on extended capabilities of finite differences integrator tool TMAP7.

Special attention is paid to radiation parameters defining inputs acting on transport magnitudes. These inputs have been analysed by using ionizing/damage radiation transport tools such as MCNPX/SRIM.

VW operational scenarios are discussed with special attention being paid to the ITER design assumptions for the values of H-species source terms (neutrals and implanted) in the ECRH system. The available material transport database with and without radiation is discussed and taken as reference for this parametric exercise. Permeation fluxes through base materials are shown to be below DRG limits established for ITER.

ACCURACY ASSESSMENT OF THE IN-BED CALORIMETRY EMPLOYED IN ITER SDS

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The SDS(Storage & Deliver System) constitutes the fuel cycle of ITER. Major task of the SDS is to store the hydrogen isotopes and deliver them to the fuel injection system. The hydrogen isotopes are stored in a ZrCo bed in the form of ZrCo hydride confined in the primary and secondary containment within the glove box of N2 atmosphere. The tritium inventory of the bed is determined from the decay heat of the tritium without removing the content from the bed. The decay heat is measured by an in-bed calorimetry: circulating He through the ZrCo bed and measuring the resultant temperature increase of the He flow. Heat lost by several heat transfer mechanisms, mostly thermal radiation, will affect the accuracy of the calorimetry. This paper, taking such heat loss into account, presents a quantitative prediction of the heat transfer rates and resulting temperature increase of the calorimetric He flow subject to the temperature and mass flow condition using the CFX code. The accuracy of the calorimetry is assessed based on the prediction.

PS4-2025

SAFE HANDLING EXPERIENCES OF TRITIUM STORAGE BEDS

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In fusion reactor, a few kg of tritium or more will be stored in hydride beds. In case of ITER, a ZrCo bed for storing 100 g of tritium has been designed with self-accountancy ability and more than 30 of beds will be used in the fuel storage & delivery system. In order to enhance total safety of fusion facility with tritium, the safe design and operation of storage beds will be one of the most important points. Concerning the safety design, the effect of tritium decay, such as decay heat transfer and He-3 behavior, is a key issue with the normal protection of over temperature, over pressure and leak for a metal-hydride bed. Concerning on the safety operation, procedure of hydrogenation-dehydrogenation cycle is a key issue under the requirements of the storage system in the fusion facility. Also, emergency performances, such as a rapid hydrogen recovery and loss of normal cooling function, should be investigated.

In the Tritium Process Lab. in Japan Atomic Energy Agency, many tritium storage beds with ZrCo and U have been used with/without self-accountancy ability, and the safe handling experiences have been accumulated for almost 20 years. In this paper, the experiences concerning the above key issues and failures summarized, and the enhancement of safety is discussed for future tritium storage system.

INITIAL REFERENCE DESIGN OF ZRCO HYDRIDE BEDS FOR ITER

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To develop ZrCo hydride bed for the storage and delivery system (SDS) and the long-term storage system (LTS) in the ITER, detail design of the internal structural arrangement and the vessel of SDS and LTS beds was implemented by using the ASME VIII Div. 1 pressure vessel design code. The SDS bed is composed of primary and secondary vessels. The primary vessel contains ~ 3.5kg ZrCo, a centralized heater tube, outer heater on the surface of primary vessel, heat transfer fins, and He loop for in-bed tritium measurement. Space between the two vessels provides a guard vacuum zone, and it contains thermal reflectors to reduce heat transfer to the secondary vessel, and secondary vessel surface temperature. Three independent He gas flow paths are provided for removal of the tritium decay heat, for rapid cooling of the ZrCo powder packed section for in-bed tritium inventory measurement in short time (8-24h), and for purging of the tritium permeated from the primary vessel into the guard vacuum zone. The general structure of the LTS bed is similar to that of the SDS bed. In order to validate an initial reference of ZrCo hydride beds, the following heat analysis was performed: (i) heating performance of ZrCo hydride packed bed during delivery operation, (ii) cooling performance (decay heat removal, etc) to reduce the ZrCo hydride temperature during recovery and storage operation, (iii) thermal shielding performance for reduction of outer vessel surface temperature. The present paper provides the results of heat analysis and discuss the effects of endothermic and exothermic reaction (ZrCo + 1/2T2 = ZrCoT + 80.495kJ/mol (T2)), and of cooling using He gas.

PS4-2027

DEVELOPMENT OF A NEW DETECTION SYSTEM FOR MONITORING HIGH LEVEL TRITIATED WATER

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In ITER a huge amount of elemental tritium is loaded as a fusion fuel, and a given amount of tritium is injected into the reactor core through a storage-delivery system of the fuel circulation system. However, the exhaust gases from the divertor contain various kinds of tritium species such as QT, QTO, CnQmT (Q=H, D) as well as original fuel particles and He.
Extraction and recovery of tritium from such tritium-containing species are one of important issues from viewpoints of safety and economy of tritium. Especially, processing of tritiated water (HTO and DTO) becomes important since high level tritiated water is necessarily generated in the fuel circulation system. From this viewpoint, it is indispensable to develop a technique for non-destructive measurements in a processing system of tritiated water.
Although a liquid scintillation counter is widely used for evaluation of tritium concentration in the water, it is difficult to apply this technique to non-destructive measurements of high level tritiated water. As a new technique for static measurement of high level tritiated water, utilization of beta-ray-induced X-ray spectrometry (BIXS) has been recently proposed by the author. In this technique a single type X-ray detector shielded with a thick lead wall and a metallic vial with a thin beryllium window were used to reduce the effects of natural radiations and to gain a high transmittance of X-rays. Basic principles of this technique may be applicable to a flow system of tritiated water.
From this view point, an improved detection system available for a flow system of tritiated water was designed and fabricated. This system consists of two X-ray detectors without lead shielding, and they are located opposite each other to coincidently detect X-rays generated in the tritiated water. We can select either mode in single and coincident one during measurements. The effects of natural radiations were able to reduce largely by applying a coincident mode without lead shielding. Tritiated water flows in a narrow space between two detectors. The narrow space is mainly constructed by stainless steel, and two windows made by a thin beryllium plate are equipped to detect the X-rays induced by beta-rays. Performance of the improved detection system will be discussed for non-destructive measurements of high level tritiated water.

EXPERIMENTAL CONFIRMATION OF THE ITER CRYOPUMP HIGH TEMPERATURE REGENERATION SCHEME

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Forschungszentrum Karlsruhe (FZK) is developing the ITER high vacuum pumping systems for evacuation and maintenance of the required pressure levels in the torus (during burn and dwell, conditioning and leak detection), the neutral beam injectors and the cryostat vessel. All ITER high vacuum systems share the same concept of accumulative cryosorption pumping. The pumping surfaces, forced-cooled by 4.5 K supercritical helium, are coated with activated charcoal so as to be able to adsorb helium and hydrogens (H2, D2). All other gases are cryopumped by cryogenic phase transition from gaseous into the liquid/solid state. For the hydrogen processing pumps in the torus and the NBI, the maximum pumping time is given by the limitation of the maximum hydrogen inventory such that the resulting pressure in case of a loss of vacuum event and a corresponding oxy-hydrogen explosion is compatible to the design criteria of the vacuum vessel.

To limit the gas accumulation, a staggered regeneration philosophy has been adopted, which involves three different temperature levels in order to achieve high regeneration efficiencies at best availability of the pumping system. The regular regeneration step is performed at a charcoal temperature of 90K to release all hydrogen isotopomers (and helium), which are subsequently pumped out by the forevacuum pumping system. The second step at ambient temperature leads to the release of all air-like species. It has to be performed less frequently, probably over-night. Any water-like species with strong sorption bonding forces need still higher temperatures for effective desorption from the charcoal. These species comprise not only water itself but also high molecular tracers added to the water circuits in case of leak localisation and any pumped higher hydrocarbons from the plasma exhaust or. The latter in their tritiated forms may contribute significantly to the semi-permanent tritium inventory; a good knowledge of their regeneration characteristics is therefore essential for tritium inventory control.

In the TIMO test bed at FZK, a half scale pump model of the torus exhaust cryopump with fully ITER relevant cryosorbent coating has been under detailed investigation over the last years, in order to determine the required high temperature regeneration conditions (times, pressures, temperatures). To replicate the ITER conditions most neatly, multi-cycle tests have been performed, aiming to identify any poisoning effects on cryopumping that may arise in the region of high accumulated gas loads of water-likes. Furthermore, the regeneration behaviour of representative water-likes has been investigated by high resolution gas analysis. The regeneration efficiency has been assessed by comparing pumping speeds before and after the contamination of the pump with the high molecular species. This paper summarizes the experimental results and draws conclusions with respect to ITER and the regeneration frequency to be considered for the ITER operational plan.

PS4-2029

EXPERIMENTAL RESULTS TO DETERMINE THE SEPARATION PERFORFORMANCE OF THE PACKAGES USED IN CRYOGENIC DISTILLATION ISOTOPES

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The cryogenic distillation of the hydrogen isotopes represents the back-end separation process most efficient and usually used in detritiation technologies.

In our institute there were made many researches in the field of hydrogen isotopes separation. The first results were obtained based on an experimental installation – Pilot Plant for heavy water production – and in present days using a Detritiation Pilot Plant.

In our Institute, was manufactured and patented a lot of hydrophilic package for isotopic distillation of water and hydrogen and also catalysts used for isotopic exchange water-hydrogen. This items was continuously developed in order to increase the isotopic separation efficiency. The goal of this paper is to determine by experimental work the performance of the package manufactured in our institute used in the cryogenic distillation process.

To describe the separation performances was developed a mathematical model for the cryogenic distillation of the hydrogen isotopes. In order to determine the characteristics of the package, the installation was operated in the total reflux mode, for different flow rate for the liquid. There were made several experiments considering different operating conditions corresponding to various values for the refrigeration power in the column condenser.

From the bottom and the top of the distillation column there were extracted samples in order to determine the isotopic composition. Processing the experimental data obtained from these tests using the Fenske relation, we obtained the separation efficiency function of the power inside the column boiler, operating pressure and also pressure drop along the package. This efficiency is describe by the number of theoretical plates per meter (NTT/m) or by equivalent height of one theoretical plate (IETT).

ASSESSMENT OF THE GAS FLOW PATHS OF THE ITER DIVERTOR CASSETTES

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To achieve a sound and balanced design for the complete ITER primary vacuum system and to study the influential parameters, simulation calculations were performed for the gas flow through the divertor slots and along the pump ducts into the torus cryopumps. For the simulation, the ITERVAC code was employed, which covers all flow regimes from laminar flow and intermediate flow at the divertor region to molecular flow at the cryopump region.
ITERVAC represents the flow system as a network of cells which represent a part of the vacuum system with a predefined shape and length. Main parameters for the simulations are gas type, temperature, source gas pressure and pressures and pumping speeds of the pumps. The output are the maximum gas throughput and pressures of the different predefined parts of the vacuum system.
The gas passages through the divertor cassettes and torus exhaust pumping ducts form a complex network of conductances. Several asymmetries in the system due to the different shapes and conductances of the channels contribute to this complexity. A full network model of the ITER primary vacuum system including the complete path from divertor to the torus vacuum pumps was developed. A special interest in the simulation work is the torus exhaust pumping at plasma pulses. The paper shows therefore the first results of ITERVAC simulations at typical divertor pressures at plasma burning.

DESIGN OF LPCE COLUMN FOR PERFORMANCE TESTS ON TRITIUM SEPARATION WITH TLK FACILITY

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The research for the performance improvement of the Liquid Phase Chemical Exchange (LPCE) column has been carried out in Nagoya University with the collaboration of National Institute for Fusion Science (NIFS). Tritium Laboratory Karlsruhe (TLK) and NIFS have a plan to perform tritium separation experiments with the column using the facility of TLK. In-line measurements of deuterium concentration both in liquid and gas phase, implemented in the TLK experimental facility, will allow accurate characterization of separation performances.
We report a design of the column interior which was designed to fit into the existing facility dedicated for LPCE process characterization (under the limitation of the TLK facility). The experimental conditions such as liquid and gas flow-rates, temperature have been established during preliminary investigations carried out at Nagoya University. The column to be used in the TLK facility is stainless steel tube with 55 mm internal diameter and 2 m length. The tritium separation experiments are performed at 120 kPa, 343 K.
A stage-wise model was also developed to predict separative performance of the column. This model requires the channeling coefficients which are estimated through analyses and/or experiments for mass transport properties in the column. For example, the channeling coefficient which represents mass transport efficiency of the catalytic exchange is evaluated against superficial velocity of reacting gases using with a small-scale apparatus. The channeling coefficient which represents axial dispersion of the packed bed is evaluated against flow rates of water by experiment and analysis of impulse response. Analytical results with the present model present effects of the catalysis quantity and the gas-liquid ratio on separative performances of the column.

PS4-2034

EFFECTS OF THE GAS-LIQUID RATIO ON THE OPTIMAL QUANTITY OF THE CATALYST FOR THE CECE PROCESS WITH A HOMOGENEOUSLY PACKED LPCE COLUMN

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In order to improve the separative performance of a CECE (Combined Electrolysis Catalytic Exchange) process we have been carried out experimental studies on hydrogen isotope separation by a CECE process using with a LPCE (Liquid Phase Catalytic Exchange) column of trickle-type bed. Two types of trickle beds were tested in our previous study. One was the layered bed where layers of Kogel catalysts and that of Dixon gauze rings were filled in the column alternately. The other was the homogeneous bed where Kogel catalysts and Dixon gauze rings were mixed and filled in the column homogeneously. We found two major points: 1) the homogeneous bed was more efficient than the layered bed and 2) there was an optimal quantity of the catalyst for both types of beds to obtain the largest separation factor. The optimal quantity of the catalyst is affected by various factors such as catalytic activity, flow rates of fluid, temperature and so on. In this study we focused on an effect of the gas-liquid ratio. The purpose of the present study is to investigate experimentally the effect of the gas-liquid ratio on the optimal quantity of the catalyst using with a homogeneous bed.
The column is a Pyrex glass tube with 25 mm internal diameter and 60 cm length. The column is filled with Kogel catalysts (1.0 wt% Pt deposited) and Dixon gauze rings. A catalyst packed-ratio is defined as a ratio of the grain-volume of catalyst to the grain volume of the whole packings, where grain volumes mean the volume of a sphere with average diameter of the Kogel catalyst and the volume of a cylinder which has the outer shape same as a Dixon gauze ring. Hydrogen-deuterium isotope separation with the CECE equipment was performed at 101 kPa, 343 K for various values of the catalyst packed-ratio and for various values of the gas-liquid ratio. Hydrogen gas was generated by the Solid Polymer Electrolysis (SPE) electrolyzer. Maximum production rate and purity of hydrogen gas are 1 m3/h and 99.99%. The concentrations of HD or HDO in gas and liquid samples were measured using a stable isotope ratio mass spectrometer (MAT252, Thermo Finnigan) with a relative accuracy to 1 %. In the present paper we report experimental and/or analytical results for the effects of the gas-liquid ratio on the optimal quantity of the catalyst.

PS4-2035

HEAVY WATER WASTES PURIFICATION FROM TRITIUM BY CECE PROCESS

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Future fusion reactors require Isotope Separation System for tritium extracting mainly from light water. Nuclear reactors moderated by heavy water also require upgrading facility to maintain deuterium concentration in water and facility for tritium recovery. The problems of tritium removal from heavy and light water and upgrading of tritiated heavy water wastes are issue of the day as before. To date the combined electrolysis catalytic exchange (CECE) process utilizing wetproofed catalyst is the most attractive one for extracting tritium from water due to its high separation factors and near-ambient operating conditions.

The experimental industrial plant has been built in PNPI for the development of the CECE technology for hydrogen isotope separation. The process uses a LPCE column and electrolysis cells to convert water to hydrogen. The plant has been in operation about 10 years. In parallel with a development of CECE process for hydrogen isotope separation the plant is used for reprocessing tritium heavy water waste. Processing waste with the content of ~ 47 % of heavy hydrogen and 108 Bq/kg of tritium, the plant produces 99.85-99.995% heavy water and deuterium gas for science and industry. Owing to industrial demands for heavy water with reduced tritium content, the plant was modified and additional equipment and procedures were put in place to operate in the detritiation mode.

After prolonged operation campaigns it was decided to update the plant with an additional separation column connected with existing equipment. Now the main parts of plant are two 100-diameter exchange columns of 7.5 m and 6.9 m overall height correspondingly, alkaline electrolytic cells. The columns are filled with alternating layers of wetproofed catalyst developed by Mendeleev University and stainless steel spiral-prismatic packing. The first column consists of five separation sections connected through a distributor of liquid, the second column consists of three separation sections.

In operation of the updated plant in different modes, we have achieved detritiation factor 5000 at product rate 4.7 kg per day, and detritiation factor 330 at product rate 15 kg per day. The values of height equivalent to a theoretical plate (HETP) were below 20 cm at a temperature of 345 K and a pressure of 0.21 MPa.

The new LPCE column demonstrates very high separation efficiency. Prolonged testing of the plant operating at various modes adduces evidence of industrial practicability of CECE process. This process can be used for tritium and protium removal from reactor heavy water.

The paper describes some results of research and development of heavy water wastes purification from tritium.

A SIMULATION STUDY ON BURNING PROFILE TAILORING OF STEADY STATE, HIGH BOOTSTRAP CURRENT TOKAMAKS

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From the aspect of fusion burn control in steady state DEMO plant, the significant challenges are to maintain its high power burning state of ~ 3 - 5 GW without burning instability, hitherto well-known as "thermal stability", and also to keep its desired burning profile relevant with internal transport barrier (ITB) that generates high bootstrap current. The paper presents a simulation modeling of the burning stability coupled with the self-ignited fusion burn and the structure-formation of the ITB. A self-consistent simulation, including a model for improved core energy confinement, has pointed out that in the high power fusion DEMO plant there is a close, nonlinear interplay between the fusion burnup and the current source of non-inductive, ITB-generated bootstrap current. Consequently, as much distinct from usual plasma controls under simulated burning conditions with lower power (<< 1GW), the self-ignited fusion burn at a high power burning state of ~ 3 - 5 GW becomes so strongly self-organized that any of external means except fuelling can not provide the effective control of the stable fusion burn. It is also demonstrated that externally applied, inductive current perturbations can be used to control both the location and strength of ITB in a fully non-inductive tokamak discharge [1, 2]. We find that ITB structures formed with broad non-inductive current sources such as LHCD are more readily controlled than those formed by localized sources such as ECCD. The physics of the inductive current is well known. Consequently, we believe that the controllability of the ITB is generic, and does not depend on the details of the transport model (as long as they can form an ITB for sufficiently reversed magnetic shear q-profile). Through this external control of the magnetic shear profile, we can maintain the ITB strength that is otherwise prone to deteriorate when the bootstrap current increases. These distinguishing capabilities of inductive current perturbation provide steady state, advanced tokamak reactors an external means of ITB control that can be used for regulating the fusion-burn net output and spatial profile [3]. [1] O. Sauter et al., Phys. Rev. Lett. 94, 105002 (2005). [2] N. Takei, Y. Nakamura et al., to appear in Plasma Phys. Contr. Fus. (2007). [3] Y. Nakamura et al, Nucl. Fusion 46 (2006) S645-S651.

FINAL MANUFACTURE OF THE OUTER VESSEL OF THE CRYOSTAT FOR WENDELSTEIN 7-X

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WENDELSTEIN 7-X is a helical advanced stellarator which presently is under construction at the Greifswald branch of IPP. A set of 70 superconducting coils arranged in five modules provide a twisted shaped magnetic cage for the plasma and allow a steady state operation. The Operation of the magnet system at cryogenic temperatures requires a cryostat which provides the thermal protection and gives access to the plasma. The main components of the cryostat are the plasma vessel, the outer vessel, the ports and the thermal isolation.
The German company MAN DWE GmbH Deggendorf is responsible for manufacture of the plasma vessel and the outer vessel.
The main body of the outer vessel is formed by a toroidal shell with a minor diameter of 4.4 m and a major diameter of 11 m. The manufactured wall thickness is 25 mm. The outer vessel is toroidally divided into 5 modules. A module is composed by five faceted toroidal sectors. In order to allow the assembly of the plasma vessel with the ports, the magnet system and support structure, each module of the outer vessel is made and delivered in separate lower and upper semi-shells. The outer vessel includes 524 domes; some of them have multiple openings. Thus there is a total of 549 domes.
Extensive finite element analyses were completed to define the wall thickness of the domes and the welding seams between the main body and the domes.
The sectors of each module were formed by rolling. Five sectors were welded together to form one module shell. The vacuum tightness of the welds was tested by a helium leak test. Precise cutting of the holes for the ports was performed by plasma cutting and following milling. The welding of the domes was performed under control of a laser positioning system. Finally a pre-assembly was performed. The contours of the semi-shells and the positions of the domes were measured by laser tracking system and were well within the given narrow tolerances. The semi-shells of the first module were manufactured by the end of 2006. After installation of thermal isolation the semi-shells will be delivered to IPP.
This paper describes the manufacturing technology and explains the solutions to the problems with the large openings.

MECHANICAL EXPERIMENTS ABOUT PENDULUM SUPPORT OF VACUUM VESSEL W7-X

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At the Max-Planck-Institut für Plasmaphysik (IPP) the new fusion experiment WENDELSTEIN 7-X (W7-X) is under construction.
Its toroidal plasma (major diameter of 11 m) is enclosed by the Vacuum Vessel (VV). The superconducting magnetic system is located around the VV and generate the Cryo-Vacuum in the Outer Vessel (OV). All vertical forces from the VV must be led through the Cryo-Vacuum and the Outer Vessel to the machine base. This is the main function of the VV support.
In addition the VV support has to also to allow horizontal movements. It is necessary because of thermal expansion of the VV (up to 20 mm) and because of horizontal movement of the VV during adjustment procedures. In order to take the vertical forces from the VV and allow the described movements, pendulum supports were introduced in the VV design. The paper will shortly describe their function mode.
Furthermore the paper will also include the description of the test campaign at the University of Rostock. To verify the supporting system all specified functions and parameters of the pendulum supports were proved.
A test frame with hydraulic equipment was built, where the pendulums were tested on the scale 1:1 (max. length 1800 mm) and loaded with the same forces as expected during operation. (130 kN). The tests have found out the required horizontal forces for all types of pendulums. On its base the overall friction factors in lubricated and non-lubricated status were determined.
In addition for the semispherical bearing, different tolerances, sphere materials, and hardness were investigated. The determined horizontal forces are to be used when designing adjacent components of W7-X. All in all the results show that a maintenance-free safe function of the VV supports is warranted.

FINAL DESIGN AND MANUFACTURING OF THE CRYOLEGS TO W7-X-SUPERCONDUCTING COIL MAGNET AND SUPPORT SYSTEM

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One of the most complicate task during assembly of the W7-X is the installation of the superconductive coil system. The coils are supported by a circular support structure, the coil support structure (CSS). The entire magnet system is enclosed between the outer vessel and plasma vessel in ultra high vacuum at 4K.
The CSS carries all 70 coils. The CSS is designed as closed ring made of 5 modules (10 half modules). The supporting of the CSS takes place via 10 cryolegs acting on the machine base (MB). The cryolegs are static highly loaded components. These legs are transfer elements between the cold parts embedded in the cryostat (outer vessel) and the machine-base at ambient temperature. The cryolegs take over five substantial tasks:

- Transmission of high vertical and horizontal forces (max Fv = 1000 kN, max Fh = 156 kN)

- Thermal insulator between the cold parts (CSS / coil system) embedded in the cryostat and the components working at ambient temperature (cryoleg components connected to the machine base)

- Compensation of different thermal expansions between the CSS at 4K and the machine base at ambient temperature

- Compensation of building and assembly tolerances between the CSS and the outer vessel

- Vertical and horizontal adjustment of the CSS on the machine-base

All components of the cryolegs, except the insulator socket, are made of stainless steel 1.4429 316LN (yield point Rp 0,2: > 900 MPa with 4 K, elongation at fracture: > 25 %, Young's modulus : > 190 GPa at 4 K, cobalt content < 2000 ppm).
Due to its high mechanical requirements of the insulator tube a separate R&D is necessary for the development, building and test prototypes and the following manufacturing of the real parts. In parallel the operability of the bearing components is to be proven by specified test. The design of the cryolegs is described in the present paper together with their calculations and technical characteristics. Additionally the investigations for the employment of the insulator tube and the test of endurance of the cryoleg bearings are presented.

K-0003 (Invited)

STRATEGY TO DEVELOP REACTOR STRUCTURAL AND PLASMA FACING MATERIALS

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Based on conceptual power plant studies like the European Power Plant Conceptual Study (PPCS), the associated performance goals and requirements for structural and plasma facing materials (PFMs) will be briefly reviewed. In all studies, the challenging environments require materials highly resistant to a combination of high heat fluxes, irradiation damage, thermo-mechanical stresses and chemical erosion or corrosion. While for ITER or first generation fission reactor designs the maximum damage level achieved by any structural material is on the order of a few displacements per atom (dpa), the structural materials of DEMO reactors will operate up to damage levels approaching 150-200 dpa. Even more, fusion neutrons will generate high production rates of He and hydrogen isotopes enhancing sensitively irradiation embrittlement. The major strategy elements of the near and long-term research activities of the international materials community will be discussed.

With respect to structural materials, an updated road map will be shown with individual development paths for the major material classes (i) reduced activation high-performance ferritic/martensitic steels and nano-scaled oxide dispersion strengthened ferritic steels, (ii) V alloys, (iii) SiCf/SiC composites and (iv) refractory materials. For the timely availability of materials design data for fusion power plants, the international fusion materials community is working on a broad based IEA coordinated R&D programme, including (i) neutron irradiation programmes, presently up 30 dpa in mixed spectrum and 70 dpa in fast breeder reactors, (ii) advanced manufacturing and joining technologies, and (iii) multi-scale modelling for the understanding of materials properties.

In contrast to structural materials, ITER will yield important information on the operation of PFMs under plasma conditions which are relevant to those in a fusion power plant. Tungsten-based materials are regarded as main candidate for the protection of the reactor structural materials. Besides the plasma interaction, important issues are the embrittlement under neutron irradiation and the re-crystallization upon overheating. Development paths for tungsten-based materials will be shown and its perspectives outlined.

It becomes more and more obvious that computational materials science, combined with experimental validation of the simulations, could significantly advance the development of suitable materials and therefore might also be taken into account in fusion road maps. Although at present materials simulation algorithms are still far from a robust prediction of real materials behaviour when subject to high irradiation, they might be capable in future to allow reliable extrapolations at least in specific cases.

Finally, the role of the small specimen test technology and the International Fusion Material Irradiation Facility (IFMIF) as an indispensable caterpillar for providing main cornerstones of a Demo oriented materials database will be outlined.

P4-0001 (Invited)

STATUS OF IFMIF DESIGN AND R&D

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The research in fusion by magnetic confinement is more and more oriented towards the design of a pre-industrial reactor, hopefully for the middle of this century. Two main knowledge categories are mandatory and structure the international effort in this domain of research:

* The physics of the plasma, and in particular its behaviour in combustion, its interaction with the components facing it, and all technologies specific to fusion (superconducting magnets, very high flux components, remote handling, tritium cycle, etc.)
* The materials adequate for the plasma facing components and the structure of the machine (vacuum vessel in particular), the energy of fusion neutrons (14 MeV) and their intensity being well beyond fission neutrons.

ITER, whose construction has been decided in June 2005, is an international effort aiming at answering to the first set of questions. The knowledge and characterisation of materials for fusion will be devoted to a second installation, called IFMIF (International Fusion Materials Irradiation Facility).

IFMIF consists of a set of two parallel deuteron accelerators (40 MeV, 125 mA each, CW) bringing the beams to a liquid lithium target flowing at a velocity of about 15 m/s. The interaction between the deuterons and the lithium generates a flux of neutrons whose spectrum is rather well suited with fusion needs (main peak at 14 MeV). Three sets of test cells will host the material samples, with damage rates ranging from 50 dpa per year to a few dpa per year for the lowest part of the test facilities. The overall available volume is 8 litres.

After several conceptual phases, the Engineering Validation and Engineering Design Activities (EVEDA) are starting in the framework of a bilateral collaborative effort between the European Union and Japan, called Broader Approach.

The talk will describe the overall project, its main challenges and its organisation.

P4-0002 (Invited)

MATERIAL SYNERGISM FUSION-FISSION

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In fission and fusion reactors the common features such as operating temperatures and neutron exposures will have the greatest impact on materials performance and component lifetimes. Developing fast neutron irradiation resisting materials is a common issue for both fission and fusion reactors. The high neutron flux levels in both these systems lead to unique materials problems like void swelling, irradiation creep and helium embitterment. Both fission and fusion rely on ferritic-martensitic steels based on 9%Cr compositions for achieving the highest swelling resistance but their creep strength sharply decreases above ~ 823K. The use of oxide dispersion strengthened (ODS) alloys is envisaged to increase the operating temperature of blanket systems in the fusion reactors and fuel clad tubes in fast breeder reactors. In view of high operating temperatures, cyclic and steady load conditions and the long service life, properties like creep, low cycle fatigue,fracture toughness and creep-fatigue interaction are major considerations in the selection of structural materials and design of components for fission and fusion reactors. Currently, materials selection for fusion systems has to be based upon incomplete experimental database on mechanical properties. The usage of fairly well developed databases, in fission programmes on similar materials, is of great help in the initial design of fusion reactor components. Significant opportunities exist for sharing information on technology of irradiation testing, specimen miniaturization, advanced methods of property measurement, safe windows for metal forming, and development of common materials property data base system. Both fusion and fission programs are being directed to development of clean steels with very low trace and tramp elements, characterization of microstructure and phase stability under irradiation, assessment of irradiation creep and swelling behaviour, studies on compatibility with helium and developing fabrication and joining technologies for ferritic steels. There is also synergy in codifying mechanical design rules for high temperature structural materials. The rapid development of fusion requires a fundamental understanding and a robust predictive capability of radiation damage in materials located in high flux regions. A joint approach for solving material problems would bring significant benefits, including the acceleration of development of both areas.

P4-0003 (Invited)

STATUS OF DEVELOPMENT OF FUNCTIONAL MATERIALS WITH PERSPECTIVE ON BEYOND ITER

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Any engineering system is composed of functional materials as well as of structural materials, and more advanced systems tend to demand a more important and versatile role to functional materials. In nuclear fusion systems, examples of principle functional materials will be breeders and neutron multipliers for tritium production, coatings on structural materials for corrosion-resistance, MHD-loss-reduction and control of tritium permeation, thermal insertions for heat transport control, and optical and electrical materials for plasma and environmental diagnostics. For incarnation of a nuclear fusion power plant, namely DEMO, development of the functional materials with appropriate properties is essential.
A role of functional materials depends strongly on a specific design of DEMO, namely designs of systems for tritium-breeding, system-cooling and heat-transfer. In the framework of ITER project, development of tritium blanket modules (TBM) is underway. Also, in parallel with the ITER project, a complemental program called the BA (Broader Approach) is launched for realization of a DEMO nuclear fusion reactor in an appropriate time schedule, where key issues of the nuclear fusion engineering needed for the DEMO will be studied under EU/Japan collaboration. In the meantime, technologies and materials needed for diagnostics and control of burning plasma are extensively discussed under the framework of International Tokamak Physics Activity (ITPA).
The present paper will review a present status of development of functional materials from views of internationally coordinated activities based on fundamental aspects of the DEMO demands as well as from views of activities based on specific but currently dominant DEMO designs. Examples of functional materials reviewed here will be solid breeders, beryllium and beryllium alloys, coating layers on structural materials, thermal inserts, and some electrical and optical materials.

OXIDATION BEHAVIOR OF SIC/SIC COMPOSITES FOR HELIUM COOLED SOLID BREEDER BLANKET

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SiC/SiC composite is one of the candidate structural materials for a fusion reactor blanket because of its low induced radioactivity, excellent high temperature mechanical properties and excellent radiation resistance. Helium (He) gas cooled blanket (HCSB) has been considered as one of the blanket design concepts using the SiC/SiC composite for relatively high temperature plant operation. Chemical stability, especially an oxidation resistance, is a key issue to be solved for the HCSB structural material because He gas in the HCSB might include partial oxygen.

The desired strength of SiC/SiC composite can be given by an optimized interface layer between the fiber and matrix (F/M interface). In order to improve its mechanical properties, several advanced F/M interfaces such as an SiC/C multilayer (ML) and a porous SiC have been developed. However, SiC/SiC composites have a possibility of F/M interface degradation by oxidation at fusion reactor operating condition, for example by the reaction of C + O2 -> CO2.

The purpose of this study is to evaluate the oxidation behavior of SiC/SiC composites with conventional pyrolitic carbon interface (PyC-SiC/SiC) and advanced multilayer interface (ML-SiC/SiC) in He+O2 environments at 1273K.

The SiC/SiC composites used in this work were fabricated at ORNL. Reinforced SiC fiber was 1D Hi-Nicalon Type-S fiber. SiC matrix was beta-SiC fabricated by a forced chemical vapor infiltration (FCVI) process. The average thickness of F/M interface (pyrolitic carbon and SiC/C multilayer) was 1000nm. Samples were machined into 2mm x1.5mm x 2mm blocks and the surface of them was mechanically polished.

Oxidation tests were carried out using a thermal gravimetric analysis (TGA) equipment. Mixtures of He with 1500ppm O2 were used. Samples were heated from room temperature to the test temperature (1273K) at 40K/min. and then held at the test temperature for 100h. Experimental conditions included 100sccm flow rate and system pressure of 1 atm. Surface microstructural analysis was performed before and after the oxidation test with a scanning electron microscope (SEM), an optical microscope and an electron probe micro analysis (EPMA).

The PyC-SiC/SiC showed a significant weight loss up to 20h and almost no change after 20h. While the ML-SiC/SiC showed a slight weight gain up to 100h. SEM observation indicated the carbon interface recession in the PyC-SiC/SiC and almost no change of interface in the ML-SiC/SiC. In both samples, the formation of silica (SiO2) was locally observed in the fiber and matrix. Resulting from these weight change behavior and surface analysis, significant weight loss of the PyC-SiC/SiC might occur due to the oxidation and recession of the carbon interface. On the other hand, slight weight gain of the ML-SiC/SiC might occur due to the formation of silica.

RHODIUM-COATED MIRRORS DEPOSITED BY MAGNETRON SPUTTERING FOR FUSION APPLICATIONS

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Metallic mirrors will be essential components of all optical spectroscopy and imaging systems for plasma diagnostics that will be used in ITER. Any change in the mirror performance, in particular its reflectivity, will influence the quality and reliability of detected signals. Due to its high reflectivity in the visible wavelength range and its low sputtering yield, rhodium may be a good candidate material for first mirrors in ITER. However, the very high price of the raw material calls for using it in the form of a film deposited onto metallic substrates. The development of a reliable technique for the preparation of high reflectivity rhodium films is therefore of the highest importance. Rhodium layers with thicknesses of up to 2 µm were produced on different relevant substrates (Mo, Stainless Steel, Cu) by magnetron sputtering. Produced films exhibit a low roughness, crystallite size of about 10 nm with a dense columnar structure. No impurities were detected on the surface after deposition. Scratch test results demonstrate that adhesion properties increase with the substrate hardness. The detailed optical characterizations of Rh coated mirrors as well as the results of erosion tests performed both under laboratory conditions and in TEXTOR will be presented in this paper.

SHORT TERM TESTS ON FIBERGLASS UNIDIRECTIONAL COMPOSITE FOR ITER PRE-COMPRESSION

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As one of the candidate materials for ITER pre-compression rings has been proposed the unidirectional fiberglass composite. In the frame of ITER pre-compression rings manufacturing and testing, material samples have been produced and tested.
In 2003 a first batch has been produced, giving results interesting, but presenting only a lower bound of the mechanical charcateristics of the material. Therefore a grip system has been developed and tested, finding the need to have longer samples (650 mm) in order to obtain reproducible and standard-conform results. This development lead to a grip system using 45° fiberglass for grips and INCONEL compression rings in order to keep in place the grips.
In 2006 the fifth batch gave results in short term tests, resulting in an ultimate tensile strength of about 2200 MPa (room temperature) and a very limited dispersion of results. Tests at 77K temperature gave a mean value gretaer than 2750 MPa with a similar dispersion.
From the above results an allowable stress value of 900 MPa can be envisaged at operating temperature. Stress relaxation tests are presently in progress.

THE ROLE OF HELIUM ON THE EMBRITTLEMENT OF RAFM STEELS

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Reduced activation ferritic/martensitic (RAFM) 7-10%Cr-WVTa steels are promising structure materials for the first wall and blanket applications in future power plants. Transmutation helium generated in the structure materials exposed to 14 MeV neutrons is believed to strongly influence material embrittlement behaviour. As fission reactors do not provide with fusion adequate He/dpa ratios, the role of helium is often studied in different simulation experiments. Experimental heats ADS2 (OPTIFER-VIII), ADS3 and ADS4 with the basic composition of EUROFER97 (9%Cr-WVTa) were doped with different contents of natural boron and separated 10B-isotope to study the effects of helium generation. In order to exclude significant differences in the microstructure, ADS2 and ADS3 were doped with 82 wppm nat. B and 83 wppm separated 10B isotope, respectively. ADS4 was doped with 1120 wppm 10B isotope.

The neutron irradiation of the reference RAFM steels (EUROFER97, F82H-mod, OPTIFER-Ia, GA3X) and the boron doped steels has been performed in the Petten High Flux Reactor up to 16.3 dpa at different irradiation temperatures between 250 and 450 °C (irradiation programme HFR IIb). The embrittlement behaviour and hardening are investigated by instrumented Charpy-V tests with subsize specimens. Boron-to-helium transformation under neutron irradiation lead to generation of 84, 432 and 5580 appm He in ADS2, ADS3 and ADS4 steels, respectively. At irradiation temperatures (Tirr) below 350°C the boron doped steels show progressive embrittlement and reduction of toughness with increasing helium amount. The analysis of the hardening vs. embrittlement behaviour at Tirr=250°C indicates that 84 appm He produced in ADS2 leads to the extra embrittlement beyond that of reference EUROFER97 steel mainly due to extra, helium induced hardening. For ADS3, however, generated helium amount of 432 appm contributes to additional embrittlement mechanisms beyond that of hardening embrittlement. At Tirr=450°C, ADS2 does not exhibit extra embrittlement beyond that of reference EUROFER97 steel. ADS3 in contrast exhibits non-vanishing extra embrittlement also at this high irradiation temperature.

Irradiation induced DBTT shift of EUROFER97 steel doped with 1120 wppm separated 10B isotope could not be quantified due to large embrittlement found in the investigated temperature range.

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RESIDUAL STRESS VARIATIONS OF SIC/SIC COMPOSITE BY HEAT TREATMENT

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In order to apply the SiC composite to fusion reactor system, among the required mechanical properties of SiC ceramics only the fracture toughness has relatively low values for fusion reactor system. One way to overcome the brittleness of the ceramics is to use ceramic fiber with controlled thermal residual stress. SiC composites are fabricated at high temperature according to the hot press (HP) process. Thus, upon cooling from the processing temperature, thermal residual stress (TRS) is arises due to the thermal expansion mismatch between the three constituents (fiber, interphase and matrix).

In this study, we are fabricate SiCf/SiC composites using a Tyranno-SA fiber that its arrangements are unidirection and two dimension woven structures. And we examined thermal residual stress between monolithic SiC matrix and SiCf/SiC composites. We used Multi-Purpose High Resolution X-ray Diffractiometer(XRD) in order to investigate thermal residual stress . And to see the microstructure used Field Emission Gun Scanning Electron Microscope System (FE-SEM).

As a results, the surface of specimen occurred microstructure formed thin oxidation film in all condition of heat-treatment. Also, the specimens occurred reacting oxidation or erosion boundary-temperature between 1200oC and 1400oC. The hardness is in inverse proportion to residual stress. The residual stress occurred compressive residual stress by condition of heat-treatment .

EFFECT OF COOLING RATE ON MECHANICAL PROPERTIES OF AGED CUCRZR

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CuCrZr is considered as the prime candidate of the heat sink material in the ITER first wall. CuCrZr which is the precipitation hardening alloy is known to be very sensitive to the heat treatment. The manufacturing condition of ITER first wall is limited by the degradation of CuCrZr. In this study, the effect of the cooling rate on the mechanical properties of the aged CuCrZr was examined. CuCrZr was solution annealed, cooled down by 4 different cooling methods and then aged at 400 to 620oC for 2 hr. Microstructure of the aged CuCrZr was observed by optical microscopy. The mechanical properties of the aged CuCrZr was evaluated by tensile test at RT and 250oC as well as Charpy impact teat at RT. Yield strength and tensile strength of the aged CuCrZr were increased with an increase of the cooling rate from air cooling to water quenching. The strength of the aged CuCrZr was highest at 440oC and then gradually decreased up to 620oC regardless of the cooling rate. Yield strength was lower than the minimum requirement value recommended by ITER design when the aging temperature was higher than 600oC.

DEVELOPMENT OF FUSION NUCLEAR TECHNOLOGIES AND THE ROLE OF MTR'S

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Fusion power plant operation will strongly depend on the economy and reliability of crucial components, such as first wall modules, tritium breeding blankets and divertors. Their operating temperature shall be high to accomplish high plant efficiency. The materials properties and component fabrication routes shall also assure long reliable operation to minimize plant outage. The components must be fabricated in large quantities based on demonstrations with a limited amount of test beds. Mock-ups and test loops will, through iteration processes, demonstrate the reliable operation under reference thermal-hydraulic conditions.

Present Materials Test Reactors, MTR's, offer fluxes relevant for large parts of the fusion major components. The mixed and fast fission spectra though is not representative for all fusion conditions. The strong point of MTR's is their ability to generate sufficient displacement damage in the materials in a relatively short time. The cores of MTR's provide sufficient space for irradiation of representative cut-outs of components to allow integrated functional and materials tests in a high flux neutron field. The MTR's are the primary test bed for structural and functional fusion relevant materials.

The MTR space and dose rates provide a valuable base line for the developments and demonstrations of fusion key components in a neutron field. In recent years the pebble bed assembly, PBA, irradiated in the HFR, Petten, has shown the feasibility of the helium-cooled concept with lithium ceramics and beryllium multiplier pebble beds. Besides the ceramic breeder concept experiments with lithium lead breeder subcomponents are continued to measure the effects of transmutation product helium on the liquid metal properties.

Similarly, activities are ongoing to perform in-pile testing of primary wall components, allowing to address fatigue type loading conditions. In the next decade 14 MeV sources such as ITER, IFMIF and maybe a volumetric source will support the crucial demonstration of components under near fusion plasma nuclear conditions. These sources have limitations in accumulated total damage (ITER) irradiation volume (IFMIF) and control. MTR's will thus continue to supply essential facts on component behaviour and materials in parallel to 14 MeV sources.

The present generation of MTR's will be closed in this and next decade because they reach their end of life. The new generation will be utilised for 4 major areas of nuclear interest: energy, science, health and environmental issues. Fusion and the next generation fission (Generation 4) power plant development will share the areas energy and science in the next decades. The design and concept of the new MTR's will centre on faster development cycles, thus higher fluxes up to 5*1018 n.m-2. Several MTR replacements in the EU are in different design stages such as the Réacteur Jules Horowitz in France and PALLAS in the Netherlands. The conceptual design of the replacement for the HFR, Petten, named PALLAS envisages a fruitful co-operation of the experimenters for advanced fission power reactor and fusion plant components.

DAMAGE STUDY FOR VARIOUS MATERIALS AT THE FIRST WALL OF A MAGNETIC FUSION REACTOR

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The main damage mechanisms at the first wall of fusion reactors under energetic fusion neutrons are gas production in the metallic lattice caused by diverse neutron induced nuclear reactions and displacement of the atoms from their lattice sites due to collisions with highly energetic fusion neutrons. These two damage types decrease the lifetime of the first wall structures. A new magnetic fusion reactor design, called APEX uses a liquid wall between fusion plasma and solid first wall to reach high neutron wall loads and eliminate the replacement of the first wall structure during the reactor's operation due to the radiation damage. In this paper, radiation damage behavior of various materials (W-5Re, V-4Cr-4Ti and SiC/SiC composite) used as the inboard and outboard first walls in the APEX blanket having the thorium molten salt, 75% LiF-23% ThF4-2% 233UF4 as a protective liquid wall was investigated. In order to evaluate the radiation damage behavior of the first wall, the tritium breeding ratio (TBR) should also be considered for (DT) fusion reactors. Therefore, tritium breeding potential of this salt with respect to investigated structural materials in the blanket was also examined. Neutron transport calculations were carried out with the aid of SCALE4.3 System by solving the Boltzmann transport equation with XSDRNPM code in 238 energy groups and S8-P3 approximation. Computations were performed with respect to the liquid wall thickness to determine effective thickness satisfying both radiation damage and tritium breeding criteria. Limits of 500 appm (atomic parts per million) and 200 dpa (displacement per atom) were considered for the helium production and the atomic displacement, respectively. On the other hand, TBR per (DT) neutron should be greater than 1.05 to maintain tritium self-sufficiency of the blanket. Radiation damage at the inner and outer first walls decreased drastically with increased liquid wall thickness exponentially. Numerical results showed that flowing wall consisting of 75% LiF-23% ThF4-2% 233UF4 with a thickness of ~35 cm and ~50 cm would be suitable to extend the lifetime of the first wall to ~30 years and supplying sufficient tritium to (DT) fusion driver for W-5Re and V-4Cr-4Ti, respectively whereas, SiC/SiC composite would require a flowing wall thickness of >60 cm to maintain damage and TBR limits.

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STRUCTURING OF TUNGSTEN BY PULSED ECM PROCESSES FOR HE-COOLED DIVERTOR APPLICATION

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A He-cooled divertor is based on components fabricated from W-alloys. Standard shaping processes e.g. turning or EDM cutting are too expensive and introduce structural defects into the work pieces. ECM as new fabrication method is known to be cost efficient and will not create damages in the surfaces. However, ECM is not used in W working due to difficulties existing in standard processes, mainly coming from the specific electrochemical properties of W. Performed electrochemical investigations showed that W can be dissolved similarly to steel alloys by applying newly adopted electrolytes which overcome passivation. These results opened the paths to examine different variants for electrochemical structuring processes of tungsten.

The two main routes which are under development towards industrial application are the M ECM and the C ECM processes exhibiting specific technological differences. M ECM is a mask-based structuring process and C-ECM, is working with a shaped 3-dimensional tool electrode. The underlying physico-chemical principle for both ECM-variants is the continuous transformation of the metal into a soluble component. Thereby, the very selective dissolution is the challenging issue which depends on interacting parameters e.g. pH-value, resist stability, distance work piece to cathode and the type of applied DC-power.

By M-ECM, applying simple constant DC-power, shaping of W was successfully demonstrated for the first time and structure depths up to 0.7 mm can be obtained easily without affecting the masks produced from a standard resist. The advantage of M-ECM is the working with an unstructured cathode, low dependency of distance work piece to cathode and simple equipment layout, however, mask stabilities, adhesion of mask to work piece or grain structure of the alloy showed more pronounced and sometimes also limiting effects. Best results were obtained with grain orientation parallel to etching directions. Also the effect of different resists types on structuring accuracy will be discussed.

The C-ECM process works with a shaped tool electrode agitated by a micrometer step-motor and the surfaces of the work pieces are uncoated. This fact implies that localized physico-chemical effects have to be used for structuring. The investigations showed clearly that strong distance effects are present concerning structuring accuracy and that distance control has to work in the low 10 µm range. The next parameter with a strong effect on structuring behavior is the shape and duration of the applied DC current pulses. In the frequency range 1 to 105 Hz a nearly linear dependency was found for the structuring accuracy. Meanwhile aspect ratios of up to 10 can be shaped with this method and depths up to roughly 1 mm can be achieved. Additional interacting process parameters are e.g. flow control of the electrolyte in the working gap and step rates of electrode motion. Beyond successful microstructuring of W parts for the cooling fingers in divertor application new results show that the C-ECM process can also be applied to parts of macroscopic shape e.g. thimbles. For both application routes metallurgical testing showed that ECM produces smooth surfaces without dangerous microcracks The progress in ECM technology will surely affect the possibility to use W as structural material in future.

DEVELOPMENT OF A PROCESS FOR TUNGSTEN COMPONENTS

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For future fusion power plants, a high amount of tungsten components will have to be fabricated for the plasma facing components. Due to the materials properties of tungsten (high melting point, high hardness, high sputtering resistance, high thermal conductivity) a cost-effective production of components is needed. Powder Injection moulding (PIM) as a method to prepare complex shaped parts in large-scale serious production at a comparably low price might be a suitable way to solve this task.

Thus, a complete PIM production process including feedstock formulation, injection moulding, debinding and thermal consolidation was developed for tungsten components. To improve the PIM performance various W powders with a particle size between 0,7 µm FSSS and 3.0 µm FSSS were tested regarding feedstock viscosity and sintering activity. This investigation was performed before and after powder deagglomeration by jet milling. Deagglomeration was found to be necessary for low feedstock viscosity at a powder loading larger 50 Vol.% as well as for enhanced sinter ability. For a low feedstock viscosity, a powder particle size larger 2,0 µm FSSS was found to be superior, while the sintering activity increases with decreasing particle size. Thus further experiments were done, applying a powder with a particle size slightly below (1,2 µm FSSS) and above (2,5 µm FSSS) the superior particle size of 2,0 µm FSSS.

By powder injection moulding laboratory samples like e.g. cooling promoter for a Helium cooled divertor according the Slot Array design [1] were replicated applying an optimized feedstock system. Density measurements of tungsten components after sintering at a temperature above 2000°C in H_2 showed a density of 19,14 g/cm³ for a powder with a particle size of 1,2 µm FSSS and of 18,37 g/cm³ for 2,5 µm FSSS, respectively. Further on the hardness of these samples was tested and a hardness comparable to recrystalized tungsten was detected (1,2 µm FSSS: 357HV10 ; 2,5 µm FSSS: 324HV10; recrystallized after Lassner& Schubert [2]: 300HV30).

[1] P. Norajitra, R. Giniyatulin, N. Holstein, T. Ihli, W. Krauss, R. Kruessmann, V. Kuznetsov, I. Mazul, I. Ovchinnikov and B. Zeep, Status of He-cooled divertor development for DEMO, Fusion Engineering and Design 75–79 (2005) 307–311.

[2] E. Lassner & W.-D. Schubert; „Tungsten: Properties, Chemistry, Technology of the Element, Alloys, and Chemical Compounds"; 1999; Kluwer Academic / Plenum Publishers, New York; ISBN 0-306-15053-4

MICROSTRUCTURE INVESTIGATION OF BRONZE /STEEL BRAZED JOINTS PROPOSED FOR HHF COMPONENTS OF ITER MANUFACTURING

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Brazing is considered as one of the perspective option of high heat flux components of ITER manufacturing. CuCrZr bronze, austenitic steel AISI 321-type and PM-17 brazed material (Ni-Mn-Fe-Si-Sn-B alloy) were used for the development of brazing technology.
Two type of brazing have been studied within the framework of recent investigation:
- HIP assisted brazing,
- Furnace assisted brazing (with uniaxial compression loading).
For the hydrostatic pressing (HIP) the brazed components were pressed out for about 175MPa during 2.5 hrs at the temperature 1035-1040oC. For the furnace assisted brazing all components were inserted into the sealed can, vacuumed and heated up to brazing temperature ~ 950oC. Fast cooling and ageing heat treatment (500oC & 4 hrs) were applied to provide high strength of CuCrZr bronze.
Microsections of specimens cut from the joints were studied on optical microscope and SEM. The microstructure, distribution of alloying elements of base metals and of brazed material components were studied in the joints. Results of these studies are discussed in the paper.

PS5-2001

NUMERICAL ANALYSIS OF FREE SURFACE INSTABILITIES IN THE IFMIF LITHIUM TARGET

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The International Fusion Materials Facility (IFMIF) facility uses a high speed (10-20 m/s) Lithium (Li) jet flow as a target for two 40 MeV / 125 mA deuteron beams. The major function of the Li target is to provide a stable Li jet for the production of an intense neutron flux. For the understanding the lithium jet behaviour and elimination of the free-surface flow instabilities a detailed analysis of the Li jet flow is necessary.

Different kinds of instability mechanisms in the liquid jet flow have been evaluated and classified based on analytical and experimental data. Numerical investigations of the target free surface flow have been performed.

Previous numerical investigations have shown in principle the suitability of CFD code Star-CD for the simulation of the Li-target flow. The main objective of this study is detailed numerical analysis of instabilities in the Li-jet flow caused by boundary layer relaxation near the nozzle exit, transition to the turbulence flow and back wall curvature. A number of CFD models are developed to investigate the formation of instabilities on the target surface. Turbulence models are validated on the experimental data.

Experimental observations have shown that the change of the nozzle geometry at the outlet such as a slight divergence of the nozzle surfaces or nozzle edge defects causes the flow separation and occurrence of longitudinal periodic structures on the free surface with an amplitude up to 5 mm. Target surface fluctuations of this magnitude can lead to the penetration of the deuteron beam in the target structure and cause the local overheating of the back plat. Analysis of large instabilities in the Li-target flow combined with the heat distribution in lithium depending on the free surface shape is performed in this study.

MEASUREMENT OF WAVE PATTERN DISTRIBUTION ON A LIQUID LITHIUM FLOW FOR IFMIF

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In the current design of the International Fusion Materials Irradiation Facility (IFMIF), liquid metal Lithium flow is employed as the target, and neutrons are generated by nuclear stripping reaction between Lithium and Deuterons. The Lithium flows at a speed of approximately 10m/s up to 20m/s in vacuum, and is required to remove heat from the Deuteron beams, to prevent boiling, and to generate neutron stably for a long time. This paper reports experimental study on the measurement of the surface fluctuation mainly caused by waves. A simplified visualization method for measuring surface-waves flow patterns on the lithium was developed and the velocity distributions of the waves were successfully measured.
The experiment was carried out at Free Surface Test Section of Lithium Loop at Osaka University, which was build for study on the IFMIF target, and focused on the free surface behavior of the liquid Lithium flow. The test section is consists of the 1/2.5 scale target nozzle, similar to the IFMIF target with 1/2.5 scale, and straight flow channel.
In a short distance from the nozzle edge, surface waves were observed with the pattern image velocimetry technique to see distributions in velocity field. The wave patterns were tracked by photographs. Many pairs of images of waves were taken by a CCD camera with using stroboscopes. Interval times of one pair of images were adjusted and were several decades to hundred micro second, depending on the fluid velocity.
The wave patterns were tracked by an algorism called Gray Level Difference Accumulation. As the results of the experiment, distributions in surface wave were successfully measured in various velocity range of the fluid. The velocity recovery property just downstream of the edge could be measured. Experimental results from electric probe measurement are also discussed.The measurement technique is possibly be adapted to IFMIF target diagnostic and interlock systems, which are provided to maintain neutron field, integrity of target and safety in the operation.

FEATURES AND OPTIMIZATION APPROACHES OF THE ENTRANCE SECTION COOLING GAS FLOW OF THE IFMIF HIGH FLUX TEST MODULE

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The International Fusion Materials Irradiation Facility (IFMIF) is devised to contribute experimental evidence to an irradiated material properties database for candidate materials exposed to irradiation spectra and doses relevant for future fusion power reactors. Due to neutron fluxes generated by high energy deuterons reacting in a liquid lithium target, damage rates of 20-50 displacements per atom in one full power year can be achieved in steel specimens inside a volume of approximately 0.5 litres. The design of the High Flux Test Module developed at the Forschungszentrum Karslruhe (FZK) allows for maximizing the space available in the high flux neutron field for material irradiation, while at the same time allowing precise adherence of the irradiation temperature of the specimen stacks. Since enhancement of the neutron irradiation requires to locate the specimens as close as possible to the neutron source, the design proposes thin container structures (obeying mechanical constraints), and flat coolant channels between the rigs. A helium gas flow is designated to remove the heat from the rigs to keep the required irradiation temperature, which may be chosen between 250-650°C. As a result of the thin container walls and the small channel dimensions, the helium cooling gas flow is characterized by low pressure, transitional Reynolds numbers and intermediate Mach numbers.

Dedicated experimental investigations on such minichannel cooling gas flows have been conducted with the ITHEX helium loop facility. Results obtained by Laser Doppler Anemometry indicate a complex three dimensional evolution of the transitional laminar-turbulent flow field in the hydraulic entrance section. In the short cooling channels, a relevant portion of the flow alongside the rigs is influenced by this developing region. Detailed knowledge of the flow development and the resulting heat transfer coefficients is necessary to optimize the flow channel inlet design and to avoid in-homogeneities of the temperature field inside the specimen stacks, which otherwise could be caused by varying local heat transfer coefficients and mass flux redistributions (in the axial and the lateral coordinates) . Experimental results are presented and compared to numerical results obtained from calculations with the CFD code STAR-CD. Specific features of the minichannel entrance flow are identified, and conclusions are drawn for an optimized design of the entrance geometry.

ACTIVATION OF THE IFMIF PROTOTYPE DEUTERON ACCELERATOR

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The International Fusion Material Irradiation Facility (IFMIF) is projected to produce an intense neutron flux for qualifying the radiation properties of the materials for fusion power reactors. The neutrons will be generated in a Li-jet under bombardment of 40 MeV deuterons, delivered to the Li-target by two identical accelerators. They are designed as a sequence of the deuteron ion source, Low Energy Beam Transport, Radio Frequency Quadrupole (RFQ) and Drift Tube Linac (DTL) accelerators and High Energy Beam Transport finally delivering 125 mA beam current to the Li-jet target. Since such a high current accelerator is a challenge for the present technical level, the IFMIF prototype accelerator is planned to be constructed for solving relevant technology issues and demonstrating its feasibility. It will consist of a full scale injector, RFQ linac and first section of the DTL. The output 125 mA beam of 10 MeV deuterons will be investigated by diagnostics instrumentations and eventually terminated by the beam dump.
The objective of the present work is an assessment of the activation caused by deuteron beam losses in the accelerating components and by the full beam current in the beam dump during the IFMIF prototype accelerator testing period. The activation analysis was performed by the European Activation System EASY-2007 consisting of the inventory code FISPACT-2007 and the European Activation Library EAF-2007 recently extended up to 55 MeV. The deuteron activation cross sections from this library for the dominant reactions were compared with available experimental data and were validated against the thick target radioactive yields. The deuteron beam losses along the acceleration line were calculated by the Monte Carlo codes SUPERFISH and PARMILA. The transport of the decaying gamma-rays and the assessment of the radiation dose rate on the outer surfaces of the accelerator tank and beam dump have been performed by the MCNP-5 code.
The results show that activation induced by the deuteron beam losses in the accelerating components are below the legal transport and hands-on limits, but exceeds them in the case of the beam dump.

CFD CALCULATIONS ON THE IFMIF LI-JET FLUIDDYNAMICS

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IFMIF is an accelerator-based neutron source to test fusion candidate materials, in which two deuteron beams will strike a target of liquid lithium. The deuteron-lithium stripping reactions will produce the required energy neutron flux to simulate the fusion reactor irradiation.

The lithium jet must remove up to 10 MW of beam power deposited on it, so a lithium velocity as high as 20 m/s is required in the target. In addition, in the beam power deposition area, the lithium flows over a concave backwall so that the centrifugal forces avoid lithium boiling.

A stable liquid free surface is a very critical requirement of the target system, otherwise the neutron field could be altered. In this line, 1mm of amplitude has been established as the limit of lithium free surface perturbations in IFMIF present design. The experimental results of a number of water and lithium facilities together with previous fluiddynamics calculations show that the lithium free surface stability can hardly fulfill or even will exceed this design requirement. Other effects, like lithium jet thickness variation, have also been observed and predicted by calculations. Therefore, hydrodynamical stability of the lithium jet is a major issue and the possible occurrences that could affect it must be examined.

To look into these problems, a simulation of the target area has been carried out by means of a CFX 5.7 code calculation. RANS (Reynolds-Averaged Navier Stokes) CFD codes are a very useful tool to supply information of main flow parameters, but there is the necessity to validate the models supporting the results by experimental data. In addition, owing to the uncertainties associated with modelling the free surface of liquid metal with the available turbulent approaches, efforts have been devoted to support the results by means of model assessment.

The behaviour of the free surface and lithium jet thickness has been studied considering the liquid fraction volume as a first rough indicator of the surface disturbance. The heat flux to the back plate and pressures, temperatures, and velocities maps have been obtained. The occurrence of cavitation has been assessed and sensibility analysis carried out modifying some main flow parameters like velocity.

MATERIAL RESPONSES IN IFMIF CREEP-FATIGUE TESTING MACHINE

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The International Fusion Materials Irradiation Facility (IFMIF) is an accelerator driven neutron source, which is designed to perform material irradiation at conditions very close to that of future fusion reactor up to the anticipated lifetime of structural materials. Besides irradiation of material samples in the high flux test module, in situ creep-fatigue tests for structural materials in the creep-fatigue test module (CFTM) and tritium release experiments for breeder blanket materials at tritium release module (TRM) are foreseen at the medium flux test module (MFTM) of IFMIF. As it was shown previously, in situ creep-fatigue tests provide more reliable estimate of the structural material fatigue lifetime under irradiation than post irradiation tests. The present study is devoted to the detailed evaluation of material responses in the creep-fatigue machine (CFM). These are required for the design of the machine, in particular the deposited radiation energy which will be critical for the survival of the electro-mechanical components of the actuator and of the displacement transducer required to measure the strain on the gauge length of the specimen.
Neutron transport calculations were performed using McDeLicious-05 code and updated global geometry model of IFMIF. Spatial variations of displacement damage, gas and heat production rates were calculated inside creep-fatigue machine. It was shown that maximum displacement damage rate at creep-fatigue samples is about 13 dpa/fpy, while in the frame of CFM it is less than 0.9 dpa/fpy. Average heat depositions in the creep fatigue samples are 0.93 W/g for the central specimen and 0.74 W/g for the lateral specimens, whereas heat deposition in the frame of CFM is less than 0.1 W/g. Helium to dpa ratio is slightly lower than that expected for the first wall of fusion demonstration reactor DEMO. These data are discussed with respect to the materials and various components used in the CFM design.

DESIGN OF A HIGH YIELD H2+ ION SOURCE FOR COMMISSIONING OF THE IFMIF ACCELERATOR USING A ONE-DIMENSIONAL PLASMA MODEL

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It is desirable to minimise the activation of components during the commissioning phase of the IFMIF accelerator and this can be achieved by using a beam of H2+ ions in place of the D+ beam. Having the same charge to mass ratio ensures that the commissioning can be conducted at full power, provided a sufficiently intense source of H2+ ions is available. Ideally, the H2+ beam current should equal that of the IFMIF D+ beam current, 140mA, with a species fraction of 85% H2+. This paper describes a study to determine the physical processes and ion source design criteria that optimise H2+ yield.
Examination of the processes contributing to the yield of H2+ ions in a plasma indicate that extraction should occur as close to the ionisation volume as possible, i.e. a shallow plasma depth is necessary to prevent dissociative attachment. Three types of discharge are reviewed with reference to these properties. It is clear that the IFMIF reference source (the ECR source) is unsuitable due to the relatively large plasma depth. The most suitable is the volume discharge, which can be either filament or RF driven. A one-dimensional plasma model has been developed and validated against experimental measurements which span a dynamic range of three orders of magnitude. The model has been used to identify those design criteria that most strongly affect the H2+ yield. The details of the magnetic confinement field and its interaction with the emission of ionising electrons are shown to be major influences on the production and survival of the molecular ion. The code has then been used to optimise the design of a volume arc source for the production of H2+ ions.
As the emittance of a beam extracted from an ECR source is dominated by the effect of the resonance magnetic field, the beam from a volume discharge would have a lower emittance. The emittance of the H2+ beam is estimated from collisional energy transfer to be approximately 30% of that of the reference beam and it may be possible to exploit this to increase the extraction aperture. This would allow source operation at lower pressure and discharge current where an H2+ fraction of 85% could be realised simultaneously with 140mA H2+ current. If this proves not to be feasible, the total extracted current will rise to 250mA, with an H2+ fraction of 56%. The unwanted 110mA of beam current would need to be removed prior to injection into the RFQ, necessitating the inclusion of a separating magnet into the beamline. The impact of the H2+ injector on the beamline is discussed.
This work, supported by the European Communities under the contract of Association between EURATOM/UKAEA, was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

A BEAM PROFILE MONITOR FOR IFMIF

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The power density in each IFMIF deuterium beam is 500MWm^-2 at the lithium target, requiring a non-interceptive technique for monitoring the beam. The purpose of the profile monitor is two-fold: to enable the high order linear optics to be adjusted to create the flat-topped distribution at the target and to detect deviations of this distribution that may result in boiling of the liquid lithium target as an interlock signal to turn off the beams. In order to satisfy the interlock application the selected technique must be responsive on a sub microsecond timescale and this precludes the use of long integration times and data analysis.
Present day techniques used in high energy beamlines are reviewed and their suitability for application to IFMIF assessed. The most frequently used technique, Residual Gas Ionisation, would fulfil the criteria for both applications but there is no experience of operating these devices continuously. There are concerns regarding the degradation of performance with time and this is exacerbated by the high neutron flux present in the target chamber. A second, less established technique, Residual Gas Fluorescence, cannot meet the time response requirement due to a low signal to noise ratio. A related method, using the spontaneous emission from the lithium vapour in the target chamber shows more promise, although the analysis presented is rudimentary. The advantage of optical techniques lies in the possibility of positioning the sensitive detector systems at a remote location, eliminating the threat of damage by the neutron flux. The analysis of all these methods is hampered by the lack of cross section data for incident protons at the correct energy; the 40MeV IFMIF deuteron beam falls between the low energy, fusion application, and the high energy, particle accelerator application.
Two other techniques are assessed, deflection of a low energy electron beam and thermal imaging. The electron beam method uses the charge distribution of the deuteron beam to deflect a low current electron beam, a perturbation to the deuteron beam distribution resulting in a change of deflection.. Infrared imaging of liquid lithium has the advantage that it provides a two dimensional image of the beam footprint on the target. In order to satisfy the interlock requirements, the number of pixels viewed would have to be limited and some development of existing technology will be required.
The question of positioning the diagnostic is considered with reference to recent calculations of the neutron flux exiting the beam pipes in the near wall. It is clear that any diagnostic that cannot be configured with remote sensitive components must be made radiation hard. Placing the diagnostic upstream from the target chamber reduces the neutron flux but then the relationship between beam distribution at the point of measurement and at the target must be well characterised. Given that the high order optics are designed to fold the wings of the phase space distribution into the core of the beam this is a non-trivial issue. Finally some recommendations are made for future work.
This work, supported by the European Communities under the contract of Association between EURATOM/UKAEA, was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

COMPACT STELLARATOR FUSION POWER PLANT STUDIES: KEY DESIGN ISSUES AND LESSONS LEARNED

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Stellarators have many attractive features as a power plant because there is no large driven external current; they offer inherently steady state operation (with low recirculating power) and are resilient to plasma disruptions. Over the past decade or two, a number of stellarator power plants have been studied in the U.S., Japan, and Europe. These studies resulted in rather large stellarators (with a major radius of ~ 15-20 m) dictated in major part by the constraints imposed by the minimum distance between the plasma and the coils. Most recently, an integrated study of a compact stellarator (CS) power plant has been performed as part of the ARIES program. Means to reduce the device size were explored, including reducing the required minimum coil-plasma distance through neutronics optimization and developing configurations with lower plasma (or coil) aspect ratio but with "good" stellarator properties.

The study included an evolution of the machine configuration and design space through trade-offs among a large number of physics and engineering parameters subjected to design constraints, leading to the design choice of a device with a major radius of 7.75m. Our preferred power core option in a 3-field period configuration is a dual-coolant (He+Pb-17Li) ferritic-steel modular blanket concept coupled with a Brayton power cycle and a port-based maintenance scheme. In parallel with a physics effort to help determine the location and peak heat load to the divertor, we developed a helium-cooled W-alloy/FS divertor design able to accommodate 10 MW/m2. We also developed an inter-coil structure design to accommodate the electromagnetic forces within each field period while allowing for penetrations including those required for maintenance, coolant lines and supporting legs of the in-vessel components. The complex geometry of the stellarator required full 3-D analysis for a number of system and components, including CAD/MCNP analysis to estimate the overall TBR and generate the neutron wall load and plasma core radiation distributions, and 3-D analysis of the coil structure to generate the electromagnetic forces, stresses and deflections.

This paper summarizes the key engineering outcomes from the study. The design of the fusion power-core components (including the blanket, divertor and coil configuration and structure) and the results from the supporting analyses are summarized. The preferred port-based maintenance scheme and the integration of the power core within the complex geometry of a compact stellarator are briefly described. The key stellarator-specific challenges affecting the design are discussed and lessons learned from the design study are highlighted, including the impact on the power plant design and performance of the minimum plasma-coil distance, peak power density, coil design requirements and alpha loss accommodation.

P5-0002 (Invited)

PROGRESS OF DESIGN STUDIES ON AN LHD-TYPE STEADY-STATE REACTOR

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Helical Heliotrons such as the Large Helical Device (LHD) and Stellarators (H & S systems) have a high potential to realize a current-less steady-state and stable magnetic fusion energy reactor as an alternative to the tokamak DEMO-reactor. H & S systems ideally have an intrinsic property of Q = infinite. Here it is very important to remember that the understanding of the physics of 3-D toroidal magnetic confinement system is naturally extended to tokamak systems. The physics is universal among these two types of systems and the technology is common.

We present our recent results from LHD experiments and reactor studies of a next generation LHD-type DEMO Reactor called FFHR.

(1) Development of 3-D superconducting (SC) coil technology

Due to the successful results of the LHD construction from 1990 to 2007, and steady operation over 8 years from 1998 to 2007, more than 2,000 hrs/year at a high field of around 3 Tesla, we have a large enough data base to demonstrate that 3D coil technology has become the standard technology for a fusion energy reactor. LHD is the largest SC fusion device in the world, contributing to the development of the SC technology necessary for fusion research. The poloidal coils of LHD adopted a super critical forced flow cooling system and their dimensions are almost the same as the ITER toroidal coils.

(2) Extended physics understanding of high beta, high T, high n_tau_T, and steady state operation

Recent LHD experiments have demonstrated the broad and advanced capabilities of LHD as a toroidal magnetic confinement device, which are highlighted by the achievements of 5% volume averaged beta, electron and ion temperatures of 10 keV, super high density of 10E15/cc and 1 hr discharges. We plan to increase the heating power up to 35 MW, and to use deuterium gas for confinement improvement. The n_tau_T will be improved to the design nominal value of Q=0.3 within several years and ultimately would approach unity. The key issue for this is the demonstrated ability to produce super-high density plasma stably by the formation of an Internal Density Barrier (IDB), which far exceeds the tokamak Greenwald limit. IDB is formed by the careful edge control of particle and energy flux by the Local Island Divertor (LID). IDB will make it possible to pursue the new approach of a Super Dense Core Reactor (SDCR). Our results will contribute to the improvement of tokamak confinement physics.

(3) Feasability Study of Reactor Design

The SDC Reactor Concept represents a new viable scenario to build an LHD type reactor. Since the super high density core of 10E15/cc is possible, the required temperature is around 7 keV. In addition we are developing the slow reactor up scenario based on the disruption-free property of LHD. A mass based comparison of FFHR construction costs to the ITER cost database demonstrates economic viability and reasonable electricity cost. The cost of the SC helical coils does not represent a critical path.

P5-0003 (Invited)

A STELLARATOR REACTOR BASED ON THE OPTIMIZATION CRITERIA OF WENDELSTEIN 7-X

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The main advantage of stellarators is an intrinsically steady state magnetic field which, in contrast to tokamaks, is generated dominantly by external currents. Thus, unlike in a tokamak, a steady state fusion reactor based on a stellarator does not require techniques to drive large plasma currents non-inductively or to mitigate the effect of current driven instabilities. However, the disadvantage is a more complicated 3D magnetic field geometry, which needs an elaborate optimization procedure to guarantee basic reactor properties such as good magnetic surfaces, favourable equilibrium properties and magneto hydrodynamic stability at sufficiently high beta, and satisfactory thermal and fast particle confinement. This also implies a comparatively complex coil configuration with demanding production accuracy and mechanical support structure.

Stellarator optimization has led to a whole family of quasi-symmetric magnetic field configurations which overcome the problems arising from the 3D geometry. Within this family the (quasi-isodynamic) Wendelstein 7-X design achieves the smallest internal plasma currents, so that the equilibrium is provided by external magnetic field coils only and also the plasma pressure has only a weak influence. The magnetic field configuration of Wendelstein 7-X has five field periods and low magnetic shear. It is realized by 50 modular coils (10 per field period) and 20 non-planar coils for higher experimental flexibility. The whole device is designed for a discharge duration of 30 minutes which includes superconducting coils, an actively cooled divertor and an electron cyclotron resonance heating system which can deliver 10 MW over such a time period.

Various studies of a HElical Advanced Stellarator (HELIAS) reactor have been conducted already (see e.g. H. Wobig, Plasma Phys. Control. Fusion 41 (1999) A159 or C.D. Beidler et al., Nucl. Fusion 41 (2001) 1759). The HELIAS reactor is basically an extrapolation from the Wendelstein 7-X design, which in itself is based on results from the Wendelstein 7-AS stellarator, the first advanced stellarator experiment. In this paper an attempt will be made to review these reactor studies considering the experience gained from the design, construction of components and the starting assembly of Wendelstein 7-X.

WELD METAL DESIGN DATA FOR 316L(N)

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This paper extends the ITER materials properties documentations to weld metal types 316L, 19-12-2 and 16-8-2, used for welding of Type 316L(N), i.e. the structural material retained for manufacturing of ITER main components such as the vacuum vessel. The data presented include those of the low temperature (316L) and high temperature (19-12-2) grades, as well as, the more readily available grade (16-8-2).

Weld metal properties data for all three grades are collected, sorted and analyzed according to the French design and construction rules for nuclear components (RCC-MR). Particular attention is paid to the type of weld metal (e.g. wire for TIG, covered electrode for manual arc, flux wire for automatic welding), and the type and the position of welding. Design allowables are derived for each category of weld and compared with those of the base metal.

The data sheets established for each physical and mechanical properties follow the presentation established for the ITER Materials Properties Handbook (MPH). They are part of the documentation that when combined with codification and inspection documents should satisfy ITER licensing needs.

In most cases, the analyses performed, go beyond conventional analyses required in present international codes and pay attention to specific needs of ITER. These include, possible effects of exposures to high temperatures during various manufacturing stages e.g. HIPing, and effects of irradiation at low and medium temperatures. In general, it is noticed that all three weld metals satisfy the RCC-MR requirements, provided compositions and types of welds used correspond to those specified in RCC-MR.

S6-0005

CORROSION CHARACTERISTICS OF LOW ACTIVATION FERRITIC STEEL, JLF-1, IN LIQUID LITHIUM IN STATIC AND THERMAL CONVECTION CONDITIONS

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Fe-Cr-W based low activation ferritic steels are regarded as a candidate blanket structural materials for liquid blanket system in fusion reactors. Some of the blanket concepts being proposed use liquid lithium as tritium breeding materials. But the compatibility between Fe-Cr-W steels and lithium (Li) is a concern. The corrosion trend of the JLF-1(Fe-9Cr-2W-0.1C) in a static Li was investigated in the previous study. The corrosion of the JLF-1 was summarized as selective dissolution of carbon and chromium, and phase transformation from martensite to ferrite. The depth of the phase transformation was estimated as 100micrometer for JLF-1 exposed in Li for 100h at 973K, while no phase change occurred at 873 for 100h. However, the depth of the phase change could be underestimated because of solubility limit in static test condition. To estimate corrosion properties in blanket relevant flowing conditions, a corrosion test for JLF-1 in flowing Li in a thermal convection loop was carried out at 773K. In the same time, the static test at the same temperature was performed for comparison.

Coupon specimens (16x4x0.25mm) were used in both static and loop exposure. A glove box with atmospheric control, high purity lithium and argon cover gas were involved to achieve very low impurity environment. Static tests were performed at 773K for 250h in molybdenum crucible. While, a thermal convection loop test for 250h was carried out in a stainless steel loop. The temperature in hot and cold leg was 773K and 723K, respectively. The estimated velocity was approximately 0.05m/s. After the experiment, all the specimens were cleaned by water, followed by weight loss measurement. SEM/EDS were used to observe the surface and analyze the composition change. Also, cross sectional observation was carried out to examine depth dependence of microstructure.

Phase transformation occurred on JLF-1 at the hot leg (773K) and the phase transformation depth was around 10micrometer from the surface. At the cold leg, the phase change was negligible because of the effect of mass transfer. In static test, the phase transformation phenomenon was found at 973K but not at 873K. This indicates that the phase change at the hot leg in flowing conditions is more prominent than that in static conditions. Chemical analysis revealed carbon depletion in JLF-1 after high temperature exposure. This seemed to cause the phase transformation from martensite to ferrite. EDS result indicated that Cr dissolved into liquid lithium during experiment at the hot leg. The phase transformation and elements depletion resulted in a hardness decrease. The softening area is consistent with the phase transformation area. Also the corrosion rate obtained from the loop test at 773K was compared with those of the static tests.

SYSTEMATIC FUEL CYCLE SYSTEMS ENGINEERING FROM 2D FLOW DIAGRAMS TO 3D LAYOUT

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The ITER fuel cycle systems are designed to supply deuterium-tritium gas mixtures to the ITER fueling systems and to process return gas streams from the vacuum vessel forming the closed inner fuel cycle. The radioactive nature of tritium requires implementation of a multiple barrier concept in order to assure the confinement of tritium within the process equipment. Ventilation and vent detritiation systems are the part of a dynamic confinement barrier which prevents tritium releases to the environment. The ITER fuel cycle systems, ventilation and tritium confinement systems all together form a rather complex chemical plant – the ITER Tritium Plant.

Not only because of the complexity of the inner fuel cycle systems and numerous interfaces to the other systems within tritium plant but also because of the procurement sharing integrated planning is required. Interfaces management, configuration control and systems integration requires proper CAD tools and Project Data Management systems.

CATIA V4 has been used in the past in ITER for 3D planning. However, only today's version of the software allows linking of the primarily 2D Pipe and Instrumentation Diagrams (P&IDs) into detailed 3D design and layout. The capabilities of the software were demonstrated through proof of principle activities in the ITER CAD office, eventually leading to the decision to deploy CATIA V5 Equipment and Systems (E&S) as general purpose single CAD tool for the design and integration of the ITER electrical, fluid and mechanical systems.

In order to meet engineering requirements of ITER the CATIA V5 E&S project structure and project resources have been established starting from systems classifications, followed by the implementation of the applicable industrial standards, specifications and systems elements libraries into the Project Resources Management (PRM). Catalogues for the piping parts, piping specifications and standards specific for the design of the tritium processing systems and tritium confinement systems will assure implementation of the Design Guidelines and Quality Requirements for the Tritium Plant systems including the standardization of the equipment and design.

The paper will describe the CATIA V5 E&S project structure, the procedures to develop and maintain the PRM and how the tool is employed to detail the design of Tritium Plant systems.

S6-0004

DETRITIATION STUDIES FOR JET DECOMMISSIONING

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JET is the world largest tokamak and has the capacity of operating with a tritium plasma. Three experimental campaigns, the Preliminary Tritium Experiment (0.1g T2) in 1991, the Trace Tritium Experiment (5g T2) in 2005, and the large experiment, the Deuterium-Tritium Experiment (DTE1) (100g T2) in 1997, were carried out at JET with tritium plasmas. In DTE1 about 35 grams of tritium were fed directly into the vacuum vessel, with about 30% of this tritium being retained inside the vessel.

In several years time JET will cease experimental operations and enter a decommissioning phase. In preparation for this the United Kingdom Atomic Energy Authority, the JET Operator, has been carrying out studies of various detritiation techniques. The materials which have been the subject of these studies include solid materials, such as various metals (Inconel 600 and 625, stainless steel 316L, beryllium, “oxygen-free” copper, aluminium bronze), carbon fibre composite tiles, “carbon” flakes and dust present in the vacuum vessel and also soft housekeeping materials. Liquid materials include organic liquids, such as vacuum oils and scintillation cocktails, and water. Detritiation of gas streams was also investigated. The purpose of the studies was to select and experimentally prove primary and auxiliary technologies for in-situ detritiation of in-vessel components and ex-situ detritiation of components removed from the vessel. The targets of ex-vessel detritiation were a reduction of the tritium inventory in and the rate of tritium out-gassing from the materials, and conversion, if possible, of intermediate level waste to low level waste and a reduction in volume of waste for disposal. The results of experimental trials and their potential application are presented.

S7-0001 (Invited)

IFMIF ACCELERATORS DESIGN

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The IFMIF requirement for 250 mA current of deuteron beams at a nominal energy of 40 MeV is met by means of two identical Continuous Wave (CW) 175 MHz linear accelerators running in parallel, each delivering a 125 mA, 40 MeV deuteron beam to the common target. This approach allows to stay within the current capability of present RF linac technology while providing operational redundancy in case of failure of one of the linacs. Each linac comprises a sequence of acceleration and beam transport/matching stages. The ion source generates a 140 mA deuteron beam at 95 keV. A Low Energy Beam Transport (LEBT) transfers the deuteron beam from the source to a Radio Frequency Quadrupole (RFQ) cavity. The RFQ bunches and accelerates the 125 mA beam to 5 MeV. The RFQ output beam is injected through a matching section into a Drift-Tube-Linac (DTL) where it is accelerated to the final energy of 40 MeV. In the reference design, the final acceleration stage is a conventional Alvarez type DTL with post couplers operating at room temperature. Operation of both the RFQ and the DTL at the same relatively low frequency is essential for accelerating the high current deuteron beam with low beam loss. The RF power system for the IFMIF accelerator relies on vacuum tube amplifiers operated at a power level of 1 MW and a single frequency of 175 MHz. The primary concern of the IFMIF linacs is the minimisation of beam losses, which could limit their availability and maintainability due to excessive activation of the linac and irradiation of the environment. A careful beam dynamics design is therefore needed from the source to the target to avoid the formation of particle halo that could finally be lost in the linac or transfer lines. A superconducting solution for the high-energy portion of the linac, using for example a CH-structure, as proposed by IAP Frankfurt, could offer some advantages, in particular the reduction of operational costs. Careful beam dynamics simulations and comparison tests with beam during the EVEDA phase are however necessary in order to fully assess the technical feasibility of this solution.

S7-0002 (Invited)

LATEST DESIGN OF LIQUID LITHIUM TARGET IN IFMIF

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This paper describes the latest design of liquid lithium (Li) target system in International Fusion Materials Irradiation Facility (IFMIF). IFMIF is an accelerator-driven intense neutron source for fusion reactor materials testing. The neutron flux is produced by means of two deuteron (D+) beams with a total current of 250 mA and maximum energy of 40 MeV that strike a liquid Li target circulating in a Li loop. The target system consists of a target assembly, a Li main loop and a Li purification loop. Design requirement of the Li target is to provide a stable Li jet with a wave amplitude less than 1 mm at a speed of 10 m/s to 20 m/s to handle an averaged heat flux of 1 GW/m2 under a continuous 10 MW D+ beam deposition. 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 water jet and Li loop experiments. On Li purification, a cold trap and two kinds of hot trap are applied to control impurities (T, 7Be, C, O, N) below permissible levels. Nitrogen concentration shall be controlled below 10 wppm by one of the hot trap. Tritium concentration shall be controlled below 1 wppm by an yttrium hot trap. Other requirements are assurance of safety with respect to the Li hazard and tritium release from the Li loop and achievement of system availability of more than 95% during plant lifetime. To maintain reliable continuous operation, various diagnostics on surface waves, Li thickness, etc. are attached to the target assembly. The target assembly needs to be exchanged at least every 11 months. Among the target assembly, a back-plate made of RAFM is located in the most severe region of neutron irradiation (50 dpa/y).
Therefore, two design options of replaceable back wall ("Cut and weld" and "Bayonet" type) and their remote handling systems are under investigation.

S7-0003 (Invited)

IFMIF TARGET AND TEST CELL – DESIGN AND INTEGRATION

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The International Fusion Material Irradiation Facility (IFMIF) aims at the qualification of appropriate materials for a Demonstration Fusion Power Plant (DEMO) to a fluence of up to 150dpa (displacement per atom) at a DEMO typical neutron spectrum. It comprises two accelerators each providing a deuteron beam with 125 mA and 40 MeV. The deuterons strike a lithium target and create via stripping reactions neutrons. The neutrons are mainly forward directed into the High-Flux-Test-Module (HFTM). The Medium Flux-Test-Modules (MFTM) and the Low-Flux-Test-Modules (LFTM) are arranged in beam direction behind. In the HFTM a damage rate in steel of more than 20 dpa/fpy (displacement per atome per full power year) will be provide in a volume of 0.5 litre. The neutron spectrum is prone to produce helium and tritium in steel like in the first wall of a DEMO reactor. The Medium-Flux-Test-Modules are designed for creep fatigues in situ and tritium release test. The test modules are cooled with helium.

The target is a lithium jet with a free surface towards the deuteron beams. The jet follows a concave curved so called back wall. Centrifugal forces increase the static pressure, which prevents lithium boiling at the beam tube pressure and the power release of 10 MW due to the deuteron beams.

The target and Test Cell (TTC) houses the target and the test modules as well as the lithium supply tubes and a quench tank into which the lithium splashes after the target. The lithium containing components have a temperature of 250 to 350 °C. Nuclear reactions mainly in beam direction contribute to heat releases in TTC components. The TTC is filled with a noble gas with almost atmospheric pressure. Natural convection transfers heat to the walls but also mitigates temperature peaks.

The Forschungszentrum Karlsruhe (FZK) has developed or validated tools for:
- The extended Monte Carlo Code McDeLicious for calculations of the neutron source term, dpa rates in the material specimens, activation and shielding and material mutations
- Pressure loss and heat transfer for helium cooling in mini channels like in the HFTM and jet cooling for the MFTM
- Code for an automatic transfer of the CAD designs to the Monte Carlo code
- Calculation for the lithium flow within the nozzle and the target

During the 6 years "Engineering Validation Engineering Design Activity" phase carried out within the frame of the Broader Approach, FZK has major responsibilities in the design of TTC, in the validation of the helium cooling systems and in various design integration activities. The presentation will give an overview on the design tools, the state of the art of the miniaturised specimens and the design integration during EVEDA.

IFMIF HIGH FLUX TEST MODULE - RECENT PROGRESS IN DESIGN AND MANUFACTURING

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The International Fusion Material Irradiation Facility (IFMIF) is an accelerator driven neutron source for irradiation tests of candidate fusion reactor materials. Two 40 MeV deuterium beams with 125 mA each strike a liquid lithium jet target, producing a high intensity neutron flux up to 55 MeV, which penetrates the adjacent test modules. Within the High Flux Test Module (HFTM) a testing volume of 0.5 litres filled by qualified small scale specimens will be irradiated at displacement rates of 20-50 dpa/fpy in structural materials. The HFTM will also provide helium and hydrogen production to dpa ratios that reflect within the uncertainties the values expected in a DEMO fusion reactor
The Forschungszentrum Karlsruhe (FZK) has developed a HFTM design which closely follows the design premise of maximising the space available for irradiation specimens in the IFMIF high flux zone and in addition allows keeping the temperature nearly constant in the rigs containing the specimen. Within the entire specimen stack the temperature deviation will be below about 15 K. The main design principles applied are (i) filling the gaps between the specimens with liquid metal, (ii) winding three separately controlled heater sections on the inner capsules and (iii) dividing the test rigs in a hot inner and a cold outer zone, which a separated by a gap filled with stagnant helium that serves as a thermal insulator. Channels between the outer covers (the cold parts) are cooled by helium gas at moderate pressure (3 bars at inlet) and temperature (50°C). 12 identical rigs holding the specimen capsules which are heated by segmented helically wound electrical heaters ensure a flexible loading scheme during IFMIF operation. Complementary analyses on nuclear, thermo-hydraulics and mechanical performance of the HFTM were performed to optimize the design. The present paper highlights the main design characteristics as well as recent progress achieved in this area. This includes the stiffening of the helium inlet duct by increased wall thickness and a gas flow baffle system, which additionally serves to redistribute the helium flow to the inlet of the HFTM test section. Former hot spots occurring in the container partitioning walls were successfully addressed by modifications in the rig surface details and cooling channel geometry. Global performance analyses of the HFTM will be presented including a first assessment of related transients expected in the operation of IFMIF.
The contribution also includes (i) recommendations for the use of container, rig and capsule materials, and (ii) a description of the fabrication routes for the entire HFTM including brazing and filling procedures which are currently under development at the Forschungszentrum Karlruhe.

S7-0006

VALIDATION OF SHUTDOWN DOSE RATE MONTE CARLO CALCULATIONS THROUGH A BENCHMARK EXPERIMENT AT JET

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In nuclear fusion reactors, DD and DT neutrons produced during operation induce the activation of the device components, thus the resulting radioactive nuclides induce high dose levels even when the machine is not operating. The problem of the activation is one of the key issues in Nuclear Fusion Technology for safe operation and maintenance and it is as more critical as the nuclear performances of the fusion machines increase. Hence in the past years many efforts have been made to develop reliable methods to predict the induced activation and the resulting shutdown dose rate.
Suitable systems of codes, data and interfaces to predict the shutdown dose rate distributions in full 3D geometry have been recently developed mainly by FZK with the Rigorous 2-Step (R2S) scheme and by ENEA with the Direct 1-Step (D1S) method. These techniques are both based on the combined use of MCNP Monte Carlo code and FISPACT inventory code, but exploit different approaches. The R2S follows a classical approach with Monte Carlo transport calculations for neutrons and decay photons in two sequential steps, whereas in the D1S method neutrons and decay gammas are transported in a single run.
Previous benchmarks performed at the Frascati Neutron Generator (FNG, Italy) and at the Fusion Neutron Source (FNS, Japan) facilities showed the effectiveness of both methods to predict dose rate, but the comparison with pre-existing, not-oriented for this purpose, dose rate measurements at JET resulted less satisfactory, therefore a dedicated benchmark experiment has been proposed. The experiment was conducted during the 2005-2007 campaign of JET in order to validate the computational methodologies in a reactor–like configuration.
Dose rate levels calculated using D1S and R2S methods were compared with experimental data collected before, during off-operational periods and at the end of 2005-2007 JET campaign in two irradiation positions: close to the vessel with high sensitivity TLDs GR-200A (natural LiF) detectors and one external position with an active detector of Geiger-Mueller type.
In this work the results of the JET benchmark experiment are presented; the impact of the nuclear activation data, coming from different evaluations, the outcome of geometrical and/or materials uncertainties and the reliability of both methods in a real fusion reactors framework are discussed as well.

STRUCTURAL MATERIAL PROPERTIES AND DIMENSIONAL STABILITY OF COMPONENTS IN FIRST WALL COMPONENT OF BREEDING BLANKET MODULE

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Various breeding blanket concepts have been proposed for ITER Test Blanket Module (TBM) and demonstrative reactors. First Wall (FW) with built-in cooling channel is a common component in the proposed blanket modules. The Hot Isostatic Pressing (HIP) method with martensitic steel tubes and plates has been developed as a practical fabrication method for the wall structure in Japan. Some mock-ups of the structure have been developed with an industrially practical process. This paper summarize the fabrication process of the first wall structure and provides the material properties of the structural material F82H and dimensional stability of the components through whole of the process.

F82H has tempered martensitic microstructure and the material should be process annealed repeatedly because of its poor deformability. The process annealing was conducted at 1083 K, which is just below Ac1 transformation temperature. This process introduced typical stretched rolling structure and ferrite/martensite dual phase structure, which lead reduction in strength. The ferrite phase is considered to be formed by decarburization in matrix corresponding to carbide coarsening at the annealing temperature. These anisotropic microstructural features, however, were successfully recovered by optimized HIP process at 1373 K, which is just below gamma-delta transition temperature of F82H.

As for dimensional stability of the components, a full-scale mockup has been developed with F82H tubes and plates. Square tubes for the cooling channel were cold–rolled with less than sub mille meter order dimensional tolerance. 11 mm x 11 mm x 1.5T x 3500L mm square tubes have been developed to fabricate the first wall without any joint in the cooling path. A bend test of F82H revealed the radius of curvature of U-shaped FW should be greater than 50 mm in order to avoid necking. The assembled components were fixed by welding without canning and the components were braced to avoid deformation during HIP process. The HIPped mockup demonstrated good accordance with a design drawing. The dimensions of wall thickness and cooling channels were to size even after HIP. According to these results, the fabrication process does not degrade the material properties and demonstrates good dimensional accuracy and stability of the FW structure.

S7-0009

DEUTERIUM RETENTION AND DESORPTION BEHAVIOR OF LITHIUM TITANATE

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In a solid blanket cooled by water, pellets of Li2TiO3 will be employed as tritium breeding material. Structure material in this blanket is low activation ferritic steel, F82H. The operation temperature is assumed to be as high as approximately 820K owing to swelling caused by neutron irradiation. Tritium produced by fusion neutrons in this breeding material has to be desorbed under a blanket operation for tritium recovery to be easy. The blanket module, however, has a spatial distribution of temperature. Thus, the tritium desorption behavior has to be known in order to make a scheme for tritium recovery.
In the present study, a solid breeding material, Li2TiO3, was irradiated by 1.7keV deuterium ions, and an amount of retained deuterium and deuterium desorption behavior were investigated using a thermal desorption. Dependence of deuterium fluence on amount of retained deuterium was also obtained. In order to examine trapping mechanisms of deuterium in Li2TiO3, similar experiments were conducted for Li and Ti.
Deuterium implanted to Li2TiO3 desorbed in forms of HD, D2, HDO and D2O. The amount of deuterium desorbed in form of HD was approximately one order of magnitude larger than those of other gas species. The desorption peak appeared at 600 K, but significant desorption up to 900 K was observed. The temperature range in the blanket is assumed from 550 K to 1200 K. These results suggest that the tritium produced in the blanket is partly not desorbed. Thus, the temperature spatial distribution in the blanket has to be controlled for the tritium to be desorbed during the operation. The desorption spectra of deuterium in Li2TiO3 were similar to those of Li. This suggests that most of implanted deuterium is trapped in form of Li-D and Li-OD. Based upon the present results, suitable design of blanket components is discussed.
This work is supported by the Grant-in-aid for Scientific Research(No. 18360439) of MEXT Japan, and partly Research Collaboration Using Fusion Facilities in JAEA.

S7-0010

ANALYSIS OF THE IN-PILE OPERATION AND PRELIMINARY RESULTS OF THE POST IRRADIATION DISMANTLING OF THE PEBBLE BED ASSEMBLIES

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The Pebble Bed Assemblies (PBA) are four tritium breeding sub scale modules, representing a segment of the European Helium Cooled Pebble Bed Test Blanket. The objective of these experiments is the study the thermomechanical behaviour of the pebble bed assemblies during irradiation. This objective will be full- filled by the analysis of changes in the in-pile temperature profiles during irradiation and the post irradiation examinition of the pebble beds in the Hot Cells.

The PBA has been irradiated in the HFR in Petten for 294 Full Power Days (FPD), to a dose of 2-3 dpa in Eurofer, and estimated lithium burnup of 2-3 %. Changes in the temperature profile during in-pile operation are indication for pebble bed creep compaction during first start up and the possible formation gas gaps between the pebble beds and the structure. During progressive irradiation the radial and axial differential temperatures within the breeder and beryllium pebble beds are evaluated. During start up of the sub sequent irradiation cycles (each 26 FPD) the temperature differences within the beryllium pebble beds show a slight increase suggesting changes in the structure of the pebble beds.

The PBA are transported from the HFR to the Hot Cell Laboratory in upright position to maintain the gas gaps between the pebble beds and Eurofer. Various microscopy preparation techniques are used to study the deformation state of the pebble beds (signs of creep compaction and sintering), formation of gas gaps between the pebble beds and structural materials and the interaction layers between eurofer-ceramic and eurofer-beryllium. In this paper first results on the Post Irradiation examination are given.

TRICICLO/PB: A COMPUTATIONAL TOOL MODELLING DYNAMIC TRITIUM TRANSFERS AT HCPB DEMO BLANKETS SYSTEMS

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The design of the cycle and the control of tritium of DEMO breeding blankets (Inner Breeding Tritium Cycle, IBTC) represent a complex and ambitious technological objective of Fusion Nuclear Technology. The IBTC design is today conceptually open to the selection and scale demonstration of tritium processing technologies and to the choice of system design operational modes and parameters.

Numerical tools modelling dynamic tritium transfers between IBTC systems based on Flow Process Diagram lay-outs support IBTC developments in many aspects serving to: (1) establish hierarchy for the IBTC design constraints and criteria, (2) to quantify on-diagram system processing technologies, (3) to fix underlying physics needed to express dynamic flux and inventories between systems, and finally (4) to make global parametric tuning and optimization of cycle parameters possible.

Among the available options, the Rankine cycle is the most conservative solution for the Power Conversion Cycle in terms of technological maturity and tritium control requirements. Optimization of Gas Cooled-High Temperature Reactor and design adaptation to DEMO primary coolant (PC) [300/500 ºC, 80bar] permit one to assess the two general diverse coolant chemistry options (HT oxidation or H2 isotopic swamping). Both options are discussed in terms of tritium control, and internal and external IBTC processing requirements for HCPB/DEMO. Permeation from the breeding ceramic into the He primary coolant and extraction of tritium by purge gas act as given inputs for the IBTC concept. Dynamic tritium transfer and radial breeding sources are inputs for actual assessments based on 2D moving-slab numerical techniques.

Ultimate tritium processing technologies performance (CPS: Coolant Purification System, TES: Tritium Extraction System from purging lines) acts as boundary IBTC design constraints. Actual limits for transient modes are discussed. The IBTC design variables concern: i) CPS system disposition in the IBTC lay-out (by-passing or not PCS), ii) use of tritium control solution at BB design level (eg. anti-permeation barrier), (iii) selection of system processing variables (ex. purge flowing velocities) and (iv) external effluents inputs for PC chemistry control.

Global tuning of a complete set of process parameters is accomplished through an ad-hoc block diagram dynamic modelling tool (TRICICLO/PB). Visual realizations of the multi-parametrical runs and optimizations for this coupled non-linear problem are given.

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PS2-2005 **Takehiko Yokomine**
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PS2-2008 **Igor Landman**
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PS2-2011 **Alice Ying**
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PS3-2004 **toshio okada**
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PS3-2005 **Yuichi OGAWA**
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