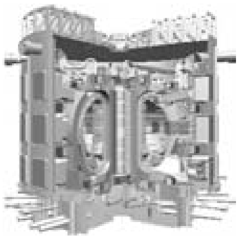

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RUSSIAN FEDERATION PARTICIPANT TEAM'S ACTIVITY IN THE AREA OF PREPARATION FOR ITER CONSTRUCTION

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Plasma physics

In the area of plasma physics the RF Participant Team (PT) is developing calculation codes in accordance with requirement of the ITER Joint Central Team, improving earlier codes, and performing calculation tasks as per signed agreements. The main directions of the RF PT activity this year are as follows:

— Integrated plasma modelling and development of ITER scenarios taking into account impurity dynamics;
— neoclassical tearing mode (NTM) control;
— study of resistive wall modes (RWMs) with models of vacuum vessel and feedback coils;
— plasma disruption simulations;
— development of plasma initiation model with a plasma equilibrium solver;
— upgrade of neutral gas transport code (EIRENE) and its coupling to B2-Eirene package (this code package is now the basic tool for studying the divertor performance in ITER);
— development of a model of high-pressure noble gas jet penetration into the plasma;
— ICRF heating and current drive modelling;
— calculation of high energy particle losses due to toroidal field ripple;
— modelling of plasma current variation in major and minor disruptions;
— development of an error field identification and correction method using the plasma response to the error fields;
— development of additional tools for the confinement database analysis code;
— analysis of the Alfven mode stability;
— plasma current and shape control.

Magnet system

Superconductor manufacture

Superconducting Nb_3Sn strands with improved properties made by the bronze process were developed and a100 kg weight qualifying batch of this strand has been produced that is now at the stage of cabling and jacketing for the preparation of TF conductor sample for testing in the SULTAN facility. The work on further improvement of the properties of the bronze process and the internal tin process Nb_3Sn strands is in progress.

Large-scale production of Nb_3Sn and NbTi superconductors with an output up to 60 t per year is continuing to be prepared. Equipment was purchased for the industrial plant chosen as the strand supplier. The manufacturing processes of initial materials and semi-finished products for superconducting strands fabrication are being adapted to and optimized for industrial conditions on new and modernized facilities.

NbTi superconducting strands were produced and cabled by RF PT which were then jacketed by the EU PT for fabrication of the conductor sample for testing in the SULTAN facility and for the fabrication of the model PF insert coil that is now ready for testing in Japan in the Central Solenoid Model Coil (CSMC) facility.

In preparation for the qualification procedure of the superconductors for the PF1/PF6 and PF5 coils, a start was made on the manufacturing of pilot batches of NbTi strands with enhanced properties.

NbTi strand for PF 1, 6 coils
Jc > 2900 A/mm^2 (5 T, 4.2 K)

Bronze processed Nb3Sn strand
Jc > 700 A/mm^2 (12 T, 4.2 K)

Internal Tin Nb3Sn strand
Jc > 800 A/mm^2 (12 T, 4.2 K)

Figure 1. NbTi and Nb_3Sn superconducting strands designed in RF PT for ITER magnet system.

Blanket system

Flexible attachment units

The experimental-industrial technology for manufacturing the flexible attachment units (FAU) for the blanket modules has been further developed. Forgings with the minimum allowances made of titanium alloy (Ti-6Al-4V) were used for FAU components making it possible to reduce steel intensity and cost of units. A set of full-scale FAUs was manufactured.

Lifetime tests of full-scale FAUs, as well as of the threaded joints used to connect these units connection with the vacuum vessel, were carried out (Figure 2). These tests confirmed the workability of the flexible attachment units, threaded joints and other design objectives.

Port limiter

The experimental-industrial technology of furnace brazing in a dynamic vacuum was developed for the port-limiter components manufacturing. The key features of this technology were handling of particular bonding fixtures and encapsulation of the port-limiter panels to be brazed. Two mid-scale mock-ups were successfully manufactured and the final products passed the quality control checks (figures 3). Full-scale mock-ups are under production.

Divertor/first wall

As from January 2005 in the area of ITER divertor/first wall development, the main directions of RF PT activity are as follows:

— Development of a semi-industrial technology for first wall assemblies and divertor armouring
  Main purpose – development of semi-industrial technology for series production of the first wall and dome-liner assemblies in the ITER construction phase.
  The main tasks at this stage are the development of semi-transparent liner elements and the manufacturing of large-scale first wall mockups using reduced cost technologies. These tasks involve design, manufacturing and tests of intermediate mockups and small-scale elements.

Figure 2. Flexible attachment unit testing

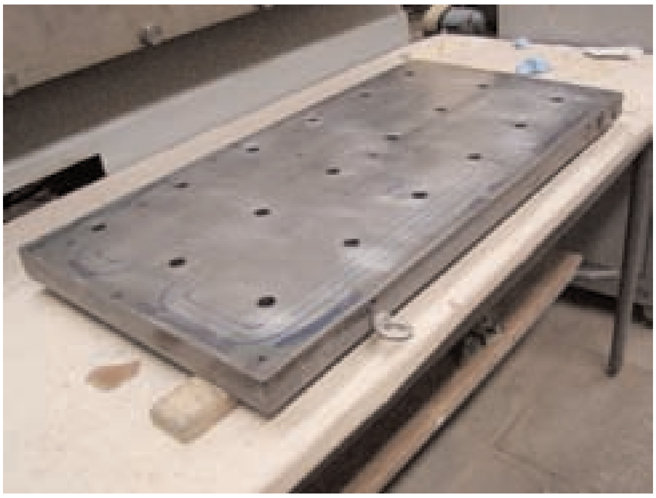


Figure 2. Flexible attachment unit testing

— Completion of the TSEFEY testing facility modernization for certification of the first wall and divertor assemblies.
Main purpose – testing equipment preparation for HHF (high heat flux) tests of first wall and divertor components in the ITER construction phase.
The main tasks at this stage are modernization of the testing facility and R&D on testing of mockups and prototypes.

— Armour materials investigation under plasma disruption and ELM events.
The main task is the development of tungsten-armoured divertor components that are reliable under plasma disruption and ELM events and to conduct tests of mockups on the plasma accelerator facility (MK-200, $q=20\ MJ/m^2$, $t=0.1\text{-}10$ mc).

— Study of hydrocarbon transport and deposition along a simulated ITER-like pumping route, which includes private flux region plasma-facing components (PFCs), warm (100-150°C) pumping duct and cryopump. Investigation of catalytic reactions of hydrocarbons on the surface in different environments (wall temperature, mixed materials, etc.). Development of co-deposited film removal technique.

— Design of private flux region PFCs. Comprehensive analysis and design of the dome-liner assembly with special attention to minimization of electromagnetic loads, the amount of activated waste, cost reduction issues and design flexibility for different schemes of hydrocarbon transport control.

Materials

R&D activity

— Materials from the divertor and first wall mock-ups prepared in EU (GlidCopAl25 IG/ SS316LN and Cu-Cr-Zr/ SS316LN *joints*) by HIP method and in RF (Cu-Cr-Zr/ SS316LN *joints*) by casting method were investigated to study the effect of complete technological cycle on strength and plasticity properties and thermal conductivity of materials (including irradiation effects).

— Investigation of the effect of in-pile bake-out on copper alloys and copper/steel joints was carried out. Specimens of GlidCopAl25 alloy and pure copper were prepared for first experiments on the effect of the irradiation-annealing–irradiation (IAI) cycle on hardening and embrittlement of copper alloys. In the RBT-6 reactor five irradiation ampoules were irradiated for 24 -1560 hours yielding accumulation of damage doses in the range of 10^{-3}-10^{-1} dpa. The investigations performed revealed that the single and double IAI cycles do not cause embrittlement in either pure copper or GlidCopAl25IG alloy in the cycles (Figure 4).

— Adaptation of the LENTA linear plasma facility for testing of ITER divertor materials.
The main task is the investigation of the ITER divertor candidate materials (C, CFC, W) in ITER-relevant divertor conditions (ELMs) and tests of prototype targets and mockups."

Beryllium

— Industrial production of beryllium tiles for the first wall continues to be prepared on the basis of recently modified manufacturing processes.

— The modelling of beryllium interaction with air and water vapour over a wide range of temperatures and pressures has been continued to confirm ITER safety margins.

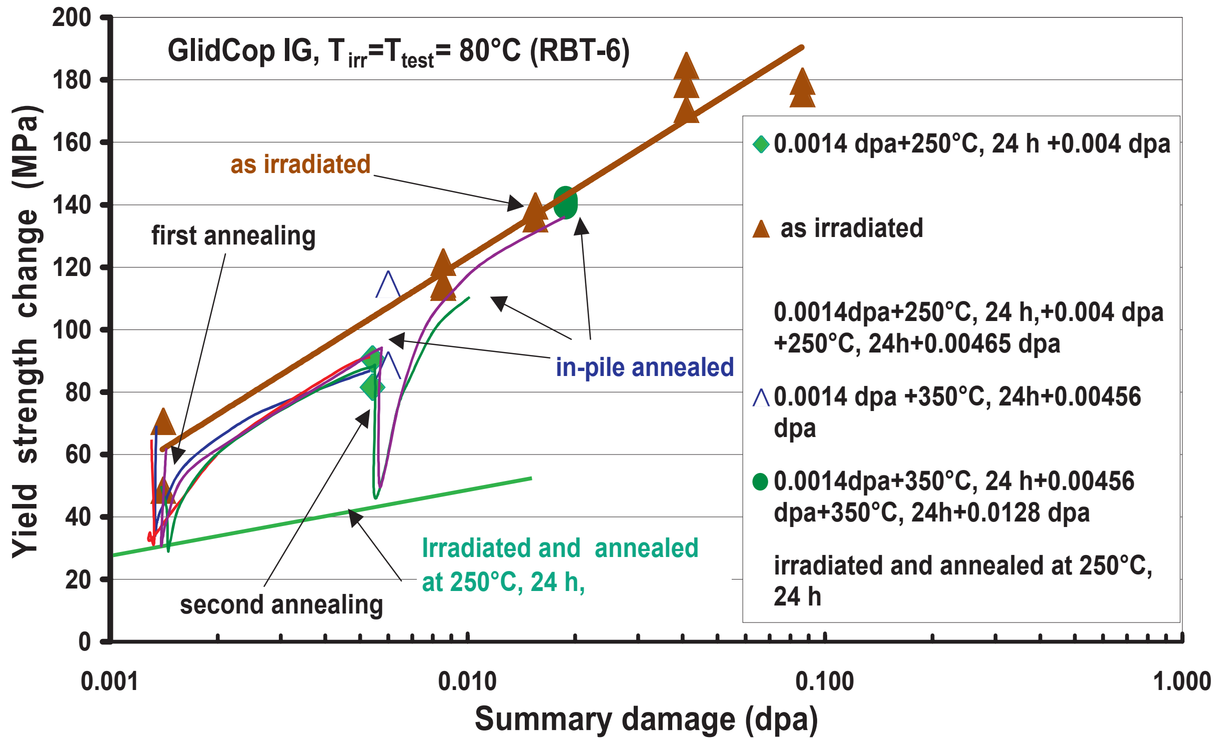


Figure 4. Change in the yield strength plotted versus neutron damage for GlidCopAl25IG alloy: as irradiated; annealed after irradiation at 250°C, 24 h; after IAI cycles.

— Samples of different grades of beryllium have been fabricated and prepared for high dose irradiation investigations in the frame of the international “HIDOBE” project. Data necessary for the ITER experimental breeding blanket modules will be collected.

Power supply

The main directions of R&D activity in the area of switching equipment for the coil power supply system at the present time are:

— Improvement of the water-cooling system of the switch contacts under 70 kA current (toroidal field fast discharge unit power supply) with the purpose of decreasing the maximum temperature of the contacts.

— Improvement of contact system of the multi-action switches for the switching network units (SNUs) with the aim to prolong the lifetime and inter-repair cycle.

— Preproduction procedure and development of the manufacturing technique of the original units and power supply components under existing industrial conditions.

Diagnostics

The RF PT is involved in the development of 17 diagnostics. In compliance with the decision on each Participant’s responsibility (2004) the Russian activity is limited to six credited (neutral particle analyzers, H alpha spectroscopy, neutron diagnostics, Thomson scattering in divertor region, reflectometry, and active spectroscopy), and several uncredited diagnostics (X-ray crystal spectrometer, gamma ray spectrometry, soft X-ray array, laser induced fluorescence, pulsed time-of-flight refractometer). The prototype diagnostic devices and their major components, such as neutron sensors, gamma-detectors, magnetic probes, ceramics, cables, optical fibres, crystal quartz, windows and mirrors were subjected to irradiation tests. Several techniques were worked out for recovery of the characteristics of diagnostics components and detectors.

Figure 5. Thermal test of the protective breaker with water-cooling under steady-state current 70 kA.

Gyrotrons

Gradually approaching the ITER requirements, Russian specialists intensively are developing 170GHz gyrotrons able to produce more than 1 MW CW output power at 50% efficiency. According to the recent test results of 170 GHz/1MW/CW gyrotrons developed in Russia within the ITER program, a power of 1.15 MW in the output Gaussian beam was attained for a pulse duration of 0.1 s. In the experiments on pulse extension, output power was not so large considering performance capabilities of the high voltage source. At an output power of 0.85-0.9 MW, the gyrotron pulse could be extended to nearly 20 s. It turned out that the transmission line elements rather than gyrotron itself were responsible for pulse limitation.

Neutral beam injector components

During the last three years the ITER NBI activity in the Russian Federation has focused on experimental validation of the ITER NBI System design (calorimeter mock-ups with swirl tube elements (STEs) made of copper and CuCrZr bronze testing), beam transmission and power deposition calculations, and preparation of the beam line components documentation and technology development with testing on prototypes.

INTERNATIONAL FUSION MATERIALS IRRADIATION FACILITY (IFMIF) PROJECT AND PROSPECTS FOR IMPLEMENTATION

by Dr. H. Matsui, IMR, Tohoku University, Sendai, Japan/JAERI Naka Establishment, Naka, Japan

Introduction

ITER is the most important milestone before the construction of a Fusion Demonstration Reactor (DEMO) but it is clear that ITER alone is not sufficient for DEMO construction. Blanket & Material development is the major technology task not covered by ITER. In addition to TBM (test blanket module) tests in ITER, material development and its database construction is essential. For this purpose, an activity toward the construction of IFMIF, the International Fusion Materials Irradiation Facility, has been pursued under an IEA Implementing Agreement since 1995 by the EU, Japan, US and Russia. In this article, the current status and future direction is briefly reported. The content of this article is mainly based on the IFMIF Comprehensive Design Report (CDR) published in January 2004 [1].

Mission and Requirements of IFMIF

The primary mission of a fusion irradiation facility will be to generate a materials irradiation database for the design, construction, licensing, and safe operation of DEMO. This will be achieved through testing and qualifying materials performance under neutron irradiation that simulates service up to the full lifetime anticipated for DEMO. Tests of blanket elements will be an important use of the facility, and will complement the tests of TBMs in ITER. The following additional missions are also recognized as important, a) advanced material development for commercial fusion reactors and b) calibration and validation of data generated from fission reactors, etc.

In order to accomplish the missions listed above, the requirements of IFMIF have been identified by the IFMIF users group as follows:

— Neutron flux/volume relation: equivalent to 2 MW/m^2 in 0.5 L volume (1MW/m^2, 4.5x10^{17} n/m^{2}s; E = 14 MeV, 3x10^{-7} dpa/s for Fe).
— Neutron spectrum:
  — should simulate the first wall neutron spectrum as closely as possible,
  — quantitative criteria are: primary recoil spectrum (PKA) and important transmutation reactions, He, H.
— Neutron fluence accumulation: DEMO-relevant fluences of 150 dpaNRT in a few years.
— Neutron flux gradient: ≤ 10% over the gauge volume

— Machine availability: 70%.
— Time structure: quasi-continuous operation.
— Good accessibility of irradiation volume for experimentation and instrumentation

Current Technical Status

IFMIF will generate neutrons with a near-fusion spectrum by hitting a flowing liquid lithium target with a pair of 40MeV deuteron beams, 125mA each. Thus, design work on IFMIF consists of four major task areas, i.e. accelerator system, target system, test cell system and design integration. During the Key Element technology Phase, KEP, from 2000 to 2002, and in the subsequent two years, significant progress was made, as described below.

In the accelerator task area, the most challenging task is to obtain availability greater than 88%. For this, long life and a stable ion source and RF power unit are essential. Extensive H+ operational experience with the Electron Cyclotron Resonance (ECR) ion source type has been obtained at CEA Saclay, with several very long runs of up to 1000 hours accumulated duration, with availability of >95% achieved [2]. Los Alamos National Laboratory (LANL) has also achieved long-term reliable operation from a similar ECR source [3].
Early in the IFMIF program, the development and testing of a 1 MW RF system was identified as the highest impact development item, Outstanding progress has been made in developing and testing a new kind of gridded tube called a diacrode. Endurance tests with a 200 MHz TH 628 diacrode have been performed during the IFMIF KEP, with over 1,047 hours at full power in the range of 1,010-1,030 kW and an availability greater than 95%. The diacrode tube has shown the capability of operating at IFMIF-relevant conditions (200 MHz and 1 MW CW).

The hydrodynamic stability of free-surface lithium jets is the key in the target system task area. The size of the IFMIF lithium jet is 250mm wide and 25mm thick, and the flow velocity is greater than 15m/s. Rather comprehensive simulation tests were conducted in JAERI using water as a simulant and the results were supported by a set of demonstration tests using Osaka University's lithium loop. In the latter tests, an existing lithium loop was modified to install a horizontal test section with a free-surface lithium channel measuring 70mm wide and 10mm thick. The stability of the jet was demonstrated up to the velocity of 14m/s. Impurity control in the lithium system is also important both from radiation safety and corrosion aspects. Significant progress was also made in controlling impurity content.

In the test cell task area, design of test assemblies is the most important subject. Test assemblies include containers of specimens to be irradiated, and their main function is to control the test environment, i.e. temperature, atmosphere, stress, etc. while maximizing the capacity for specimen accommodation. There are three test assemblies, i.e. VTA-1 (Vertical Test Assembly -1), VTA-2 and VIT(Vertical Irradiation Tube). VTA-1 contains the 0.5L high flux module where the damage rate is greater than 20dpa/year. Detailed design and thermal and stress analyses were made in the EU and Japan, showing that the specified parameters basically achieve the users requirements. Neutronics calculations also showed that irradiation parameters, e.g. helium and hydrogen production rate relative to dpa rate, are appropriate for fusion reactor material testing.

Small Specimen Test Technology (SSTT) is not a part of the design of IFMIF, but it is very important since this will determine the specimen size, shape, and number of specimens occupying the test volume. SSTT is essential for IFMIF since the irradiation test volume is far too small to accommodate standard size test specimens. Significant progress was made also in this task area, but it is still necessary to identify adequate test methods to obtain the data required for licensing purposes.

Schedule and Cost

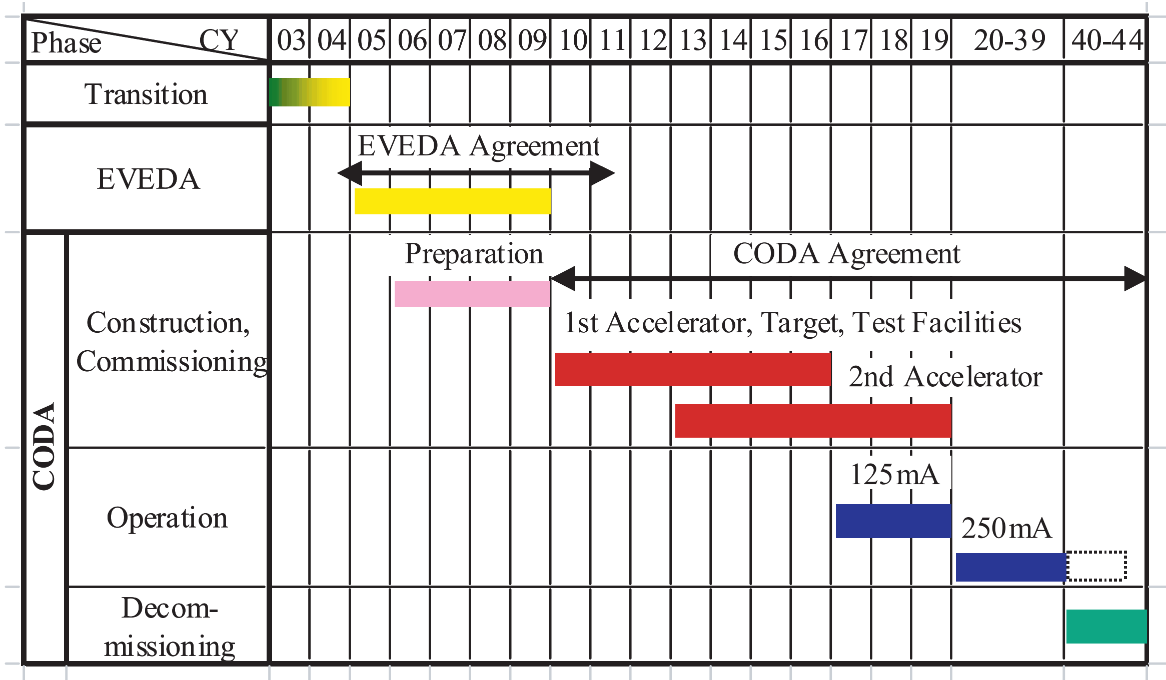
The Engineering Validation/Engineering Design Activity (EVEDA) is the phase to follow the KEP. In this phase, the following tasks are identified:

— conduct a sufficiently detailed engineering design:
    — for "call for tender" of major components, and
    — to make a decision of IFMIF construction;

— do demonstration tests in order to technically validate the engineering design;
— examine IFMIF facility operation methodology.

The detailed task list for the EVEDA has been elaborated and re-examined, but will be skipped here for reasons of space. Details are given in the CDR.

The cost for the EVEDA has been estimated as 88.4MICF, where 1 ICF is equivalent to 1 US$ as of July 2003. The cost for construction itself is 539.2MICF. The current schedule, assuming decisions on EVEDA and CODA in CY2005 and CY2010 respectively, covering the entire project is depicted in the chart below. For the construction, operation and decommissioning, a phase called CODA is identified and is planned to start in 2010. Unfortunately, the entire schedule is slipping behind because of the difficulty in the negotiation on the ITER hosting. If the Fast Track approach for fusion development is to be followed, it will be necessary to accelerate also the IFMIF construction. It appears feasible to combine the EVEDA and CODA, and to perform some of the activities in parallel to obtain the first full power neutrons earlier than is described in this chart.



* Decision on EVEDA
* Decision on CODA

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Items to be considered for inclusion in the ITER ITA Newsletter should be submitted to C. Basaldella, ITER Office, IAEA, Wagramer Strasse 5, P.O. Box 100, A-1400 Vienna, Austria, or Facsimile: +43 1 2633832, or e-mail: c.basaldella@iaea.org (phone +43 1 260026392).

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