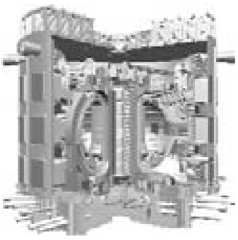

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RECENT JAPANESE PARTICIPANT TEAM (JA-PT) ACTIVITIES

By Dr. M. Mori, Leader JA-PT, JAERI

1. Introduction

The Japanese Participant Team (JA-PT) in the Japan Atomic Energy Research Institute (JAERI) has contributed to the technical activities in the ITER Transitional Arrangements (ITA) phase mostly by undertaking technical tasks under the task agreements concluded between the International Team (IT) and the JA-PT. Since the beginning of 2003, the JA-PT has undertaken 52 ITA tasks for design improvements and optimization, to provide the technical basis for narrowing down design options, or to qualify manufacturing and quality control methods, and completed 17 of them (as of April 2005). The JA-PT has not only carried out the tasks presented in the following sections, but also some tasks on buildings, diagnostics, physics, nuclear analysis, and so on.

2. Superconducting Magnet

Activities to study and demonstrate manufacturing feasibility at industrial level are in progress including full-scale trial fabrication in the following three areas according to ITER Task Agreements.

2.1 Nb_3Sn strand for TF coil

In order to prepare for mass production of Nb_3Sn strands which meet ITER requirements (critical current density (j_c) > 700 A/mm^2 for the bronze process, j_c > 800 A/mm^2 for the internal tin process, and hysteresis loss < 1000 mJ/cm^3 for ± 3 T cycle), trial fabrications of Nb_3Sn strands were performed with four potential suppliers (three for the bronze process strands and one for the internal-tin process strands). The optimum configurations of the strand cross-section were determined from measured results on the superconducting performance of trial strands with small billets, which were prepared for the study of key parameters to achieve ITER TF (toroidal field) coil requirements. Subsequently, demonstrations of production of the candidate strands with large billet size (50kg–100kg) were performed by March 2005. All strands produced in the trial fabrication will be fully qualified by the end of 2005.

2.2 TF Structure

The following products of JJ1 and nitrogen strengthened 316LN (referred to as ST316LN) for coil cases and radial plates, respectively, were produced to demonstrate mass production and mechanical properties at cryogenic temperatures.

— JJ1 rectangular forged block: 3.7 m (length) x 0.94 m (width) x 0.4 m (thickness) (as forged)
— ST 316LN rectangular forged block: 4.7 m (length) x 0.96 m (width) x 0.43 m (thickness) (as forged)
— ST 316LN (SS316LN) hot rolled plate: 1.6 m (length) x 1.8 m (width) x 0.2 m (thickness)
— ST 316LN (SS316LN) hot rolled plate: 4.8 m (length) x 1.8 m (width) x 0.14 m (thickness)

Figure 1 shows an overview of a ST316LN block at an intermediate stage in the forging process. These products will be fully qualified by using ultrasonic testing, microscopic observation, and cryogenic testing of mechanical properties by the end of August 2005. In addition, manufacturing studies of coil cases, radial plates and inter-coil structures are in progress under a collaboration with heavy industries, in order to develop a feasible manufacturing plan including segmentations, machining, welding and inspection.

2.3 CS Jacket Section

A stainless steel having a low coefficient of thermal expansion is required as a CS (central solenoid) jacket material to obtain a compressive force on winding packs in cooling down from room temperature to 4K. From this point of view, a new material, JK2LB, was selected for the CS jacket. Trial fabrications of circle-in-square JK2LB tubes involving the processes of hot extrusion followed by cold drawing were performed to confirm achievable accuracy of dimensions and available unit length of the jacket section**,** as shown in Fig. 2. A tolerance of ± 0.25 mm for both outer and inner dimensions and unit length of more than 7 m could be achieved, which satisfies the ITER requirements.

Figure 1. Overview of a ST316LN block at an intermediate stage in the forging process.

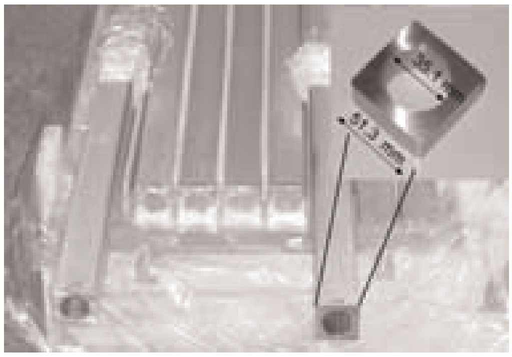


Figure 2. CS circle-in-square jacket section produced in the trial fabrication.

3.Vacuum Vessel

3.1 Structural Analysis of VV Support Structure with Multiple Flexible Plates

ITER's Vacuum Vessel (VV) is a safety-related component confining radioactive materials such as tritium and activated dust. Since the VV supports were directly connected to the TF coils in the 2001 ITER design, safety performance would have to be assured under the strong linkage of the VV and the TF coils. The JA-PT proposed an alternative VV support concept to remove the unfavourable direct linkage, whereby the VV support system with multiple flexible plates located at the bottom of the VV lower port is connected to the cryostat ring instead of the TF coil. The proposed concept has significant advantages comparing to that of the 2001 design; the TF coil can be categorized as a non-safety-related component because of the lack of direct connection of the TF coils to the VV supports.

The following results, obtained in stress analyses of the proposed VV support structure, show the feasibility of the proposed structure as an improved VV support concept.

— The maximum primary stress of the flexible plates is estimated to be 200 MPa and lower than the allowable value of 264 MPa defined in ASME Section III NF, under the off-normal load condition. The maximum stress of the connection bolt was also estimated to be lower than the allowable value.

— The maximum value of the relative displacement between VV and TF coils is found to be 15 mm, which is much less than 100 mm shown in the 2001 concept.

3.2 Fabrication of VV Partial Mock-ups

In the present design of the VV, a large number of interface structures, such as keys and housings to support the blanket modules on the VV, have been introduced as shown in Fig.3 instead of the back-plate as in the 1998 ITER design, in order to reduce the procurement cost. As a result, the number of weld joints in the VV has increased and the distance between weld joints becomes much shorter than that in the previous design, for which fabrication feasibility has already been demonstrated in the EDA R&D of the VV. The weld joints of housings for blanket supports are very close to the reinforced ribs connected between the inner and outer walls of the VV, especially in the inboard cylindrical region. Furthermore, accessibility for welding and weld inspection is extremely restricted by the highly curved inner and outer walls in the top curved region. The critical issues in the present VV design are therefore to confirm the feasibility of fabrication and quality inspection even with the expected large welding distortion and the low accessibility for welding and weld inspection.

The JA-PT is now fabricating a partial VV mock-up (shown in Figure 3), which consists of an inboard cylindrical region and a top curved region, in order to confirm the fabrication and inspection methods and in order to examine whether distortions caused by welding are within tolerances. The basic data necessary for preparing the technical specifications of the VV procurement such as weld distortions, minimum space required for welding and inspection, welding and inspection accessibility to the weld joints in the present design, and applicability of weld inspections, will be obtained through its fabrication.

4. Blanket and Divertor

For the shielding blanket, four ITER tasks have been established. One of the most important tasks is the blanket qualification task, which will be continued for several years. In the blanket design task, work has been performed to evaluate electromagnetic (EM) forces on the module, to evaluate the dynamic amplification of stress by the EM forces of the keys, and to improve the module structure design of the #10 module.

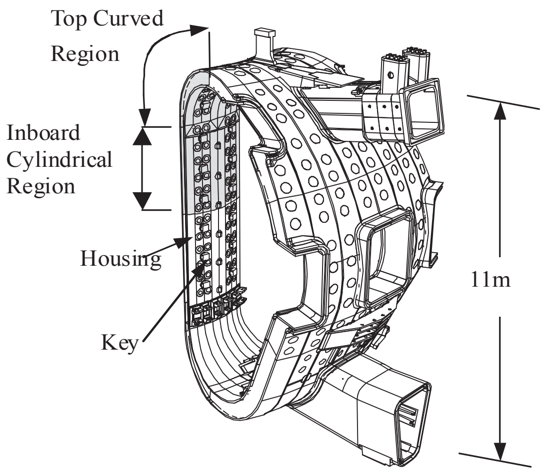


Figure 3. Fabrication of Partial Mock-up of VV.

As for the electromagnetic analysis, detailed 3-D EM analyses have been performed for a linear current decay with 40 ms quench time, based on new disruption scenarios (in ITER document, “Control Systems Design and Assessment”: CSD). Figure 4 shows a schematic picture of the analysis model of the blanket modules, divertor and half sector VV. In the preliminary analysis, the modules with only marginally acceptable loads were found. Then, structural improvements were made on these modules such as deepening the slots for the inboard module and applying a stiffer key structure for a module at the top location, in order to satisfy the design allowable limits. After the modifications, all EM forces due to eddy currents are below the design allowable limits. Also it was shown that the maximum EM force appears at the modules neighbouring the divertor region. Furthermore, the dynamic stress on the inter-modular key and the stub key have been calculated for blanket modules located near the divertor region taking into account the possible variation in initial positioning of the key and the key groove, damping effects and waveforms of EM load and thermal load. The results showed that the evaluated stress satisfied the design allowable limit.

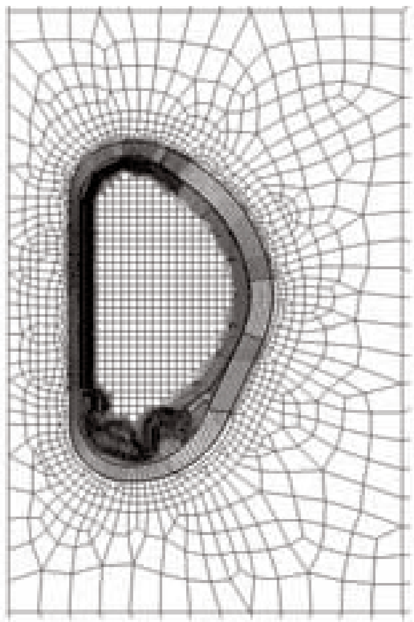


Figure 4. Schematic picture of 3-D solid model of blanket modules, divertor and half sector VV used in electromagnetic analysis of blanket modules.

For the divertor, two ITER tasks have been established. One is the development of the inboard vertical target based on annular cooling tube cooling, which has been proposed by the IT. An advantage of the annular flow concept is that it can make the vertical target more compact and provide more space for thermal expansion of the target. The JAPT has started the design work on the fabrication method of the inboard vertical target based on the annular cooling tube concept. The other task is the development of the tungsten-pin-armoured divertor with circular/rectangular cooling tubes, such as hypervapotron. In 2004, thermal cycle tests on this concept were started in the electron beam test facility in JAERI.

5. Assembly and Remote Handling

To maintain the required construction schedule for the project, the assembly operations have to be carefully evaluated. It is therefore necessary to confirm that resources (particularly the crane, and the large purpose-built tools) are not over-allocated, and that movement logistics do not adversely affect the critical path, during the assembly process. The JA-PT developed a resource-loaded schedule covering the primary stages of the ITER assembly process, using Microsoft Project scheduling software, on the basis of input data supplied by the IT, including data input, checking, debugging, etc. The identification of the critical path, loading of resources, and the study of options for achieving the required critical path, will be made using this scheduling software in the future. The JA-PT also studied the tokamak assembly scenario including metrology and assembly tolerances of the VV and TF coil.

The JA-PT updated the Blanket Test Platform (BTP) at the Naka Fusion Establishment JAERI, a legacy of the L-6 Large Project, in order to study the remote handling technique based on the latest blanket module design. The mock-up of an inboard blanket module reproduces the details of blanket interfaces such as the key and cooling pipe flange configurations based on the latest design, and also models the changing state of the gap between modules. The configuration of the remote sensing target and sensing technique will be developed using the upgraded BTP. In addition, the JA-PT started studies of maintenance and repair scenarios in the hot cell so as to clarify the repair requirements in the hot cell in detail.

6. Tritium Plant

In order to enhance the safety of the future ITER tritium plant, several investigations & evaluations were performed as follows:

(1) Oxidation performance of the detritiation system has been tested to clarify the detritiation efficiency under unusual conditions such as fire (coexistence of CO & CO_2), and it was verified that the normal oxidation efficiency of the tested detritiation system for H_2 (2%) and CH_4 (2%) in air, > 99.99% and > 99.9%, respectively, was maintained even under the coexistence of CO up to 10% and CO_2 up to 20%.
(2) Evaluation of tritium permeation through the plasma-facing components has been carried out with newly obtained experimental data. Results indicate that tritium permeation is mainly through the first wall region because of its large surface and tritium generation in the Be armour ($^{9}Be(n,t)^{7}Li$). The amount of tritium permeatiing to the secondary cooling water through the heat exchanger wall was estimated to be negligibly small (~ 0.4 GBq for 20 years).
(3) A series of irradiation tests has been carried out for the performance of organic materials in the electrolysis cells of the ITER Water Detritiation System. The cell is designed to be used for two years at a tritium concentration of 9.25 TBq/kg (530 kGy). No serious change was observed for the tensile strength and the ion exchange capacity of the membrane material (Nafion N117) of the cell was maintained up to 850 kGy of gamma irradiation.
(4) In order to investigate effective tritium removal methods from a room, intentional tritium release and its removal experiments have been carried out. Results indicate that the residual wall contamination of tritium depends on specific activity of HTO in a room air and that addition of water vapour (optimum level = 600 ppm) in the ventilation flow accelerates the removal of the residual contamination.

7. Heating and Current Drive Systems

7.1 NB system

The ITER NB system has been designed to inject 33 MW of D^0 beams from two NB systems at a beam energy of 1 MeV. To fulfill this requirement, the ITER beam source (ion source and accelerator) generates 1 MeV, 40 A (ion current density: 200 A/m^2) of D^- ion beams. Moreover, the 1 MV insulation of the accelerator should be achieved by vacuum, instead of gas to avoid radiation-induced conductivity (RIC) followed by excess heat dissipation. The objective of the present R&D is to demonstrate 1 A class H^- ion acceleration up to 1 MeV.

R&D has been carried out in the ITA period to establish the vacuum insulation technology. The vacuum insulation capability of the JAERI MeV accelerator was drastically improved by installing a large “stress ring” to reduce the electric field concentration at the triple junction between insulator, metal flange and vacuum. As a result, the JAERI VIBS (Vacuum Insulated Beam Source) succeeded in sustaining 1 MV for 8,500 s

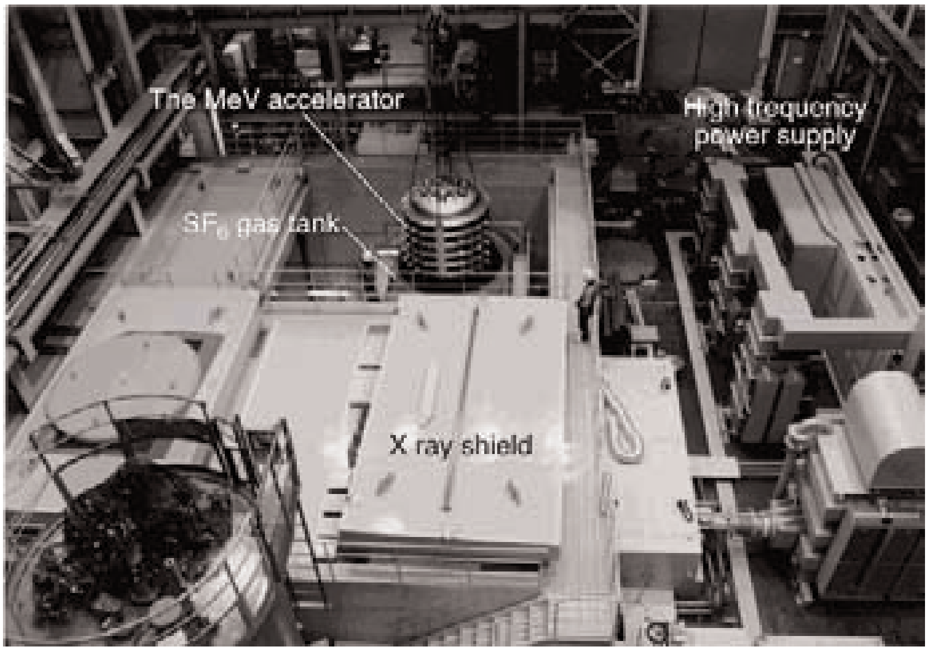


Figure 5. MeV test facility and the vacuum insulated accelerator for 1 MeV, ampere class H^- ion beam test.

continuously. At present H^- ion acceleration testing is in progress toward the ITER required current density and energy. So far, H^- ion beams of MeV range energy have been obtained with typical parameters of 836 keV, 146 A/m^2 (total ion current of 206 mA) for 0.2 s (with Cs). The pulse lengths are limited by heat handling capability of the beam dump at present. Since the KAMABOKO negative ion source being used in the present test has already demonstrated H^- ion production of 300 A/m^2 (at a beam energy of 50 keV), the current density of the ITER requirement (200 A/m^2), together with ampere level H^- ion beam current, is to be achieved soon.

7.2 RF system

ITER requires a 170 GHz high-power gyrotron system with a total power of 24 MW, for electron cyclotron heating (ECH), current drive (ECCD) and suppression of neoclassical tearing modes. Intensive development of a 1MW-170 GHz gyrotron is being carried out to satisfy the requirements of ITER. Fig.5 is a picture of RF test stand and ITER gyrotron in JAERI. Up to now, a quasi-steady-state oscillation of 100 s with 0.5 MW has been demonstrated. The temperature of major components of the gyrotron reached a steady state. In the next stage, a built-in mode converter was improved to suppress the stray radiation. In addition, constant current control of the electron beam was applied to obtain a stable oscillation. A stable current control for 1000sec was achieved. With these improvements, the further pulse extension test will be done aiming at 1MW/CW operation.

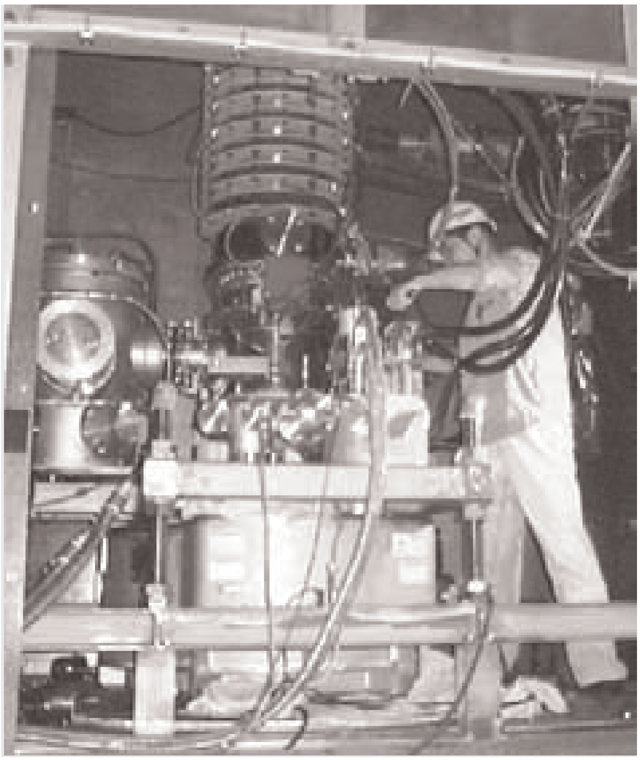


Figure 6. Picture of RF test stand and ITER gyrotron in JAERI.

Design work on the ITER equatorial EC launcher has been carried out for current drive. The launcher has three optical steerable mirrors, waveguide lines, mitre bends for each diamond windows, nuclear shields and so on. R&D of launcher components has been carried out, including neutron irradiation tests of bearings for the steering mirror and the diamond windows, and fatigue stress tests of the flexible cooling tubes for the mirror and high power RF transmission and pressure tests of the diamond windows. Nuclear analysis using the three-dimensional Monte Carlo analysis code (MCNP-4B) has been carried out. The results indicate validity of the neutron shielding design for the equatorial EC launcher.

8. Recent progress of site-dependent design study for Japanese site

A site-dependent design study has been in progress to prepare for siting of ITER in Japan. Water supply and sewage systems have been designed based on the number of workers during the construction phase. Rearrangement of the site layout from the 2001 design has been considered with the Rokkasho site conditions such as land, water and power supply etc.

Since seismic isolation is adopted for the tokamak complex building, the passage between isolated and non-isolated buildings is one of the concerns as to whether the criteria of radiation control area can be satisfied at the passage. At this point, the concept of the use of the tokamak complex building has been discussed to minimize cask transfer through the passage.

Some methods of excavation for the tokamak complex building have been studied. Methods of providing a water shield wall to deal with the ground water, an approach ramp, and their combination have been compared. By this study, the basic data has been obtained to decide the excavation method from a viewpoint of shortening the schedule after the site selection. Study of the final heat sink has also been continued.

FOURTH IAEA TECHNICAL MEETING ON NEGATIVE ION BASED NEUTRAL BEAM INJECTORS

by Dr. V. Antoni, Consorzio RFX, Associazione Euratom-ENEA sulla Fusione, Padova, Italy

The Fourth IAEA Technical Meeting (IAEA-TM) on Negative Ion Based Neutral Beam Injectors was held in Padova, Italy from 9–11 May 2005. 62 participants attended the meeting (48 from the EU, 6 from Japan, 3 from the Russian Federation, 1 from the People’s Republic of China, 3 from EFDA, and 1 from the IAEA).

The scope of the meeting was to bring together experts on the field to present the latest results and to discuss the issues on physics and technology of neutral particle injection for steady state operation in ITER (namely current density 200 A/m^2 and acceleration voltage 1MV for pulses up to 1h).

Participants to the Fourth IAEA Technical Meeting (IAEA-TM) on Negative Ion Based Neutral Beam Injectors.

Dr. G. Mank, Head of the IAEA Physics Section, said in his opening speech, that when the International Fusion Research Council, as an advising council to the IAEA, proposed to support a meeting on neutral beam injection, he thought it was a good idea, but recognized, that the last meeting, organized by the Agency must have been in the early 90's, more than 10 years ago. The restart of such an endeavour is difficult and has to be done within the existing channels of communication, research exchange and meetings. It was recognized that the neutral bean injection community is well organized (e.g. in the Coordinating Committee on Neutral Beams (CCNB)) and Mr. Mank expressed his thanks to Drs. Francesco Gnesotto and Vanni Antoni, for proposing to hold this meeting together with the CCNB. This decision helped to focus this meeting on the urgent task of negative ion beams.

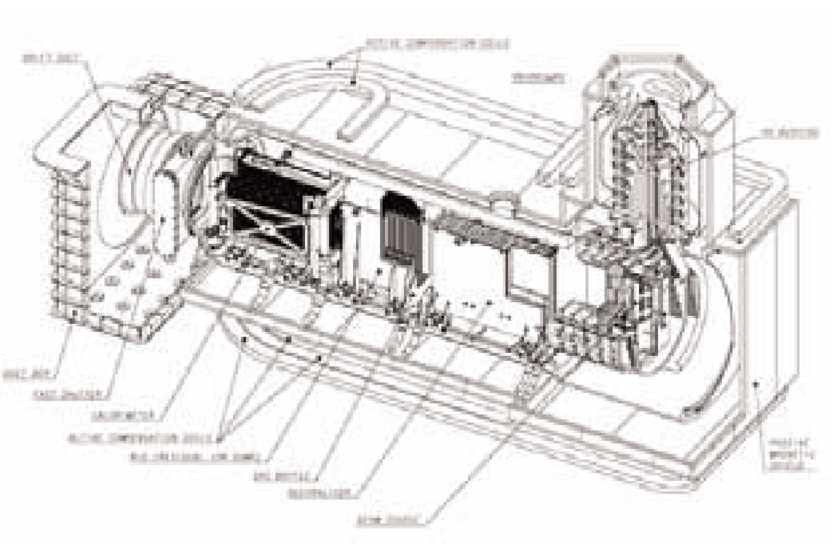
This topic attracted also the editors of the Nuclear Fusion Journal and Mr. Mank thanked Derek Stork and Ron Hemsworth for their support.

Mr. Mank continued by pointing out that the IAEA supports meetings, which are of interest to the nuclear scientific community, and in return it was of interest to the IAEA to have many Member States participate in these meetings, enabling them to gain knowledge through the distribution of the results of the meetings. Though the Padova meeting had already the size of a "mini-conference", with more than 60 participants, only 9 countries out of some 50 working on fusion participated. He was of the opinion that the work of the participants was of interest to many more countries, as research on ion sources can be supported also with lower financial resources. He hoped that the next meeting, which should not take place in another 10 years, would welcome participants from many more countries.

Mr. Mank concluded "Your contributions are important to ITER and the development of safe operating scenarios. I hope everybody will enjoy the good working atmosphere in the lovely historical city of Padova. I would like to thank Vanni and Francesco and the Italian colleagues again for inviting us and give the word to Francesco and thank you all for coming."

The topics covered during the meeting were: physics of the negative ion sources, beam operation, injector technology, and status reports and programmes.

Progress in all fields was reported. In particular, both arc and radio frequency (RF) sources have improved performance and operation reliability. RF sources have been proved to be able to meet the ITER requirements in terms of current density, operating pressure and electron co-extraction, while testing of long pulse operation in deuterium and of current density uniformity in large size sources is in progress or under preparation. Arc driven sources have demonstrated that current density and beam uniformity can be improved by optimization of magnetic filter and filament arc current control.



Scheme of the neutral beam injector for ITER.

Modeling of sources (in H and D) has confirmed that excited molecules are the precursors of negative ions and that the processes at the surfaces play a key role for operation with cesium.

New diagnostics have allowed detailed comparison of plasma properties to be carried out, showing not much difference in terms of atom, molecule and ion populations between the two sources when operating at comparable current densities and pressures.

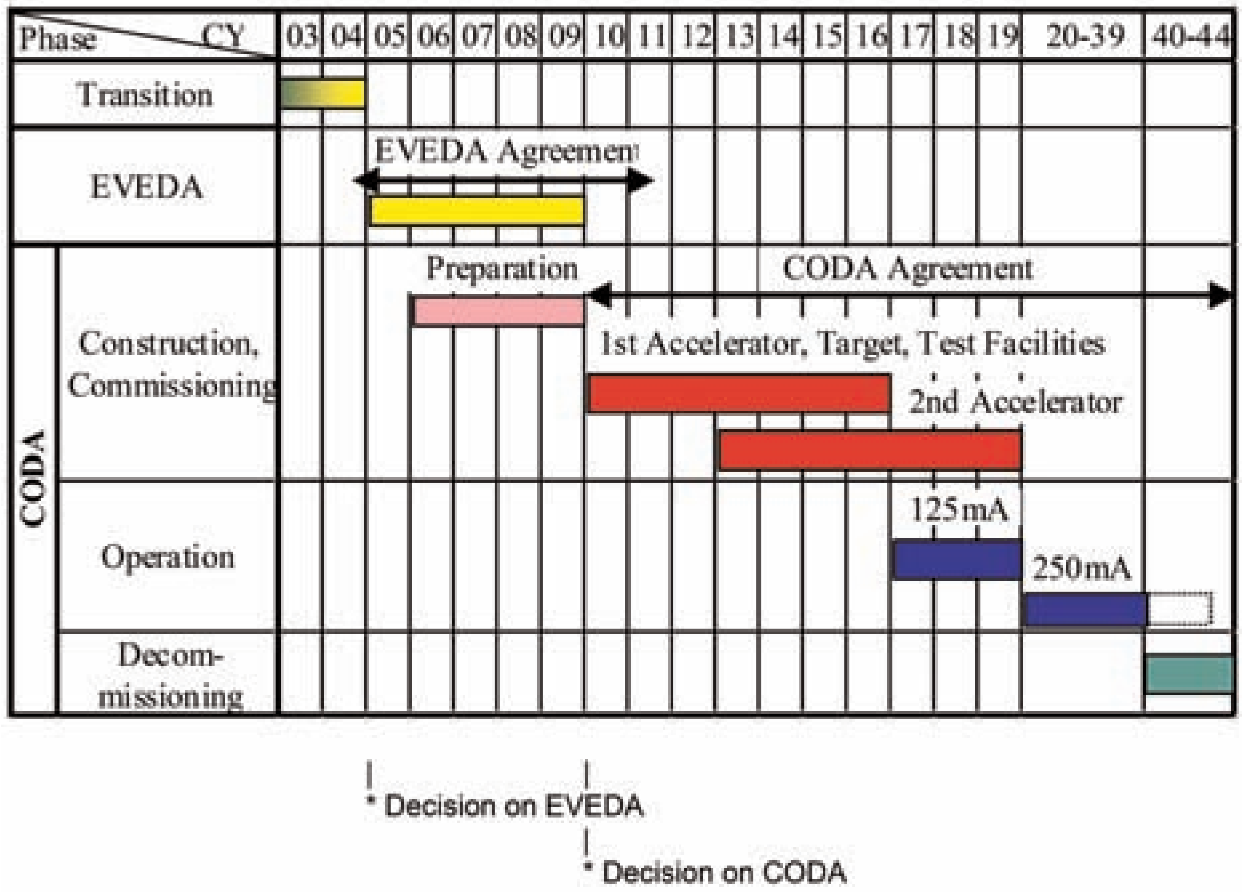
Operation at 836 kV, 146 A/ m^2 with H^- ion beams has been achieved in multi-grid accelerator systems, while improvement in optics and beam steering have been reported from single gap accelerator systems operating at 727 keV, 120 A/ m^2 with D^-. Performance of negative ion beam systems has been further improved increasing total injection power and acceleration voltage and extending the pulse length.

The outcomes of the engineering revision of the neutral beam injector design for ITER was reported focussing on the technological aspects of the realization of the bushing and of the beam line components (neutralizer, residual ion dump and calorimeter). The modifications required to adapt the injector design to a Test Facility aimed at demonstrating high voltage acceleration at ITER-relevant currents were presented and discussed. All the programmes presented by the laboratories presently involved in the negative ion neutral beam R&D confirmed the will to continue the research activity with a constructive spirit of scientific collaboration.

CORRIGENDUM

Unfortunately, in issue no. 21, April 2005, an incorrect version of the chart on page 8 was published.

We apologize for this mistake. The correct version is shown below.



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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