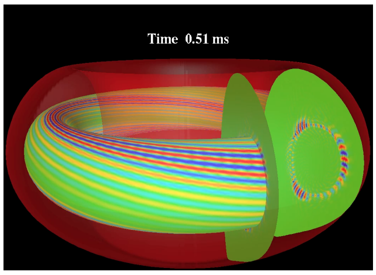
IFERC Newsletter

IFERC-N-2012-13~17 (*No. 6, 31 August, 2012*)

International Fusion Energy Research Centre, Rokkasho, Aomori 039-3212, Japan

CSC Activity

Lighthouse project: Simulation study of energetic particle driven instabilities using MEGA code



Energetic particle driven instabilities such as Alfvén eigenmodes are one of the most important issues for the ITER plasma because they lead to energetic alpha particle redistribution and losses. Computer simulation is expected to predict the stability of Alfvén eigenmodes and the associated energetic particle transport for ITER operation scenarios. This issue has been investigated in the Lighthouse Project of the IFERC-CSC Supercomputer Helios....

DEMO R&D Activity

Recent activities in EU Task 5 (Advanced Tritium Breeders)

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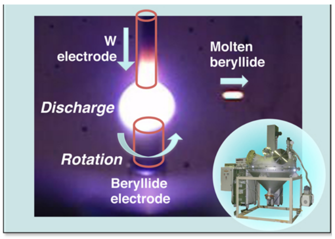
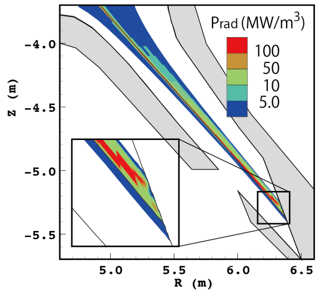
R&D activities in JA Task 4 (Advanced Neutron Multiplier)



DEMO Design Activity

Progress of power handling design toward DEMO



Meetings

Preparatory Working Group on REC has face-to-face meeting in Barcelona and Cadarache

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Lighthouse project: Simulation study of energetic particle driven instabilities using MEGA code

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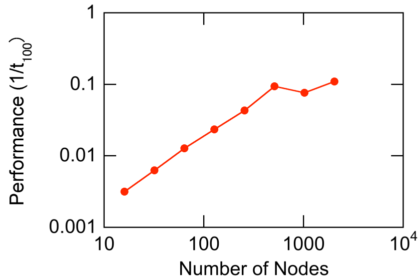

Fig. 1 Strong scaling of MEGA code vs. number of nodes on Helios

I. Performance benchmark test of MEGA code on Helios

MEGA is a hybrid simulation code for energetic particles and magnetohydrodynamics. The computational performance of a flat MPI version of MEGA code was investigated using Helios. Both the numbers of grid points and computational particles are 34 million. Figure 1 shows the computational performance vs. number of nodes on Helios. We see an excellent strong scaling up to 512 nodes with 8192 CPU cores without any optimization.

II. Alfvén eigenmode stability and energetic particle transport for ITER steady state scenario

Stability of Alfvén eigenmodes and the associated transport of energetic particles including alpha particles and beam deuterium particles were investigated for an ITER steady state operation scenario using MEGA code. The equilibrium data provided on the ITER web site is used for the simulation. The specific parameters are major radius 6.2m, minor radius 2m, and total plasma current 9MA. The particle simulation method is applied to both alpha particles and beam deuterium particles. The finite Larmor radius effects are taken into account for both the species.

It was found that toroidal Alfvén eigenmodes (TAE modes) with toroidal mode number from 12 to 22 are unstable and the saturation level of the magnetic fluctuation is 0.2% normalized by the toroidal field. Slight redistributions take place for alpha and beam deuterium particles with perturbations of beta value 0.03% and 0.01%, respectively as shown in Fig. 2. Figure 3 shows the spatial profiles of the TAE modes.

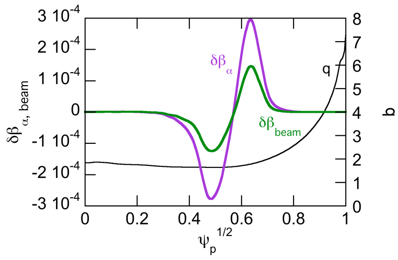

Fig. 2 Beta value perturbations for energetic alpha particles and beam deuterium particles

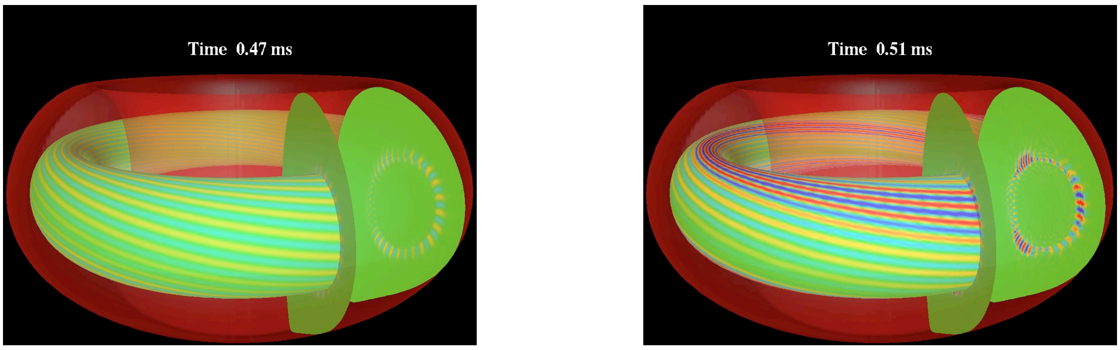

Fig. 3 Spatial profiles of TAE modes in the linearly growing phase (left) and in the saturation phase (right)

CSC helios Supercomputer number 12 in TOP500 list

On Monday, June 18 2012, the 39th TOP500 list was released at the International Supercomputing Conference in Hamburg, Germany. This is the list of the most powerful computers in the world, ordered by their performance at the Linpack benchmark. This date was important for the CSC as this was the first time that helios was represented in full. For the previous list, in November 2011, only one quarter of helios was installed and it could only achieve the 28th position. This time, it reached number 12 in the world and number 2 in Japan.

IFERC-N-2012-14 (*No. 6, 31 August, 2012*)

DEMO R&D Activity

Recent activities in EU Task 5 (Advanced Tritium Breeders)

The development, production, characterization and studies on reprocessing of advanced tritium breeders are the objectives of task 5 within the R&D activities for DEMO blankets. While in Japan the activities are focused on lithium metatitanate with excess lithium, lithium orthosilicate based materials are developed in the EU.

A melt-based process for the production of tritium breeder pebbles is preferred in the EU as it offers the possibility to reprocess the used pebbles of a fusion power plant with the same process. The used pebbles can simply be remelted and the depleted material can be replenished with lithium-6 by adding the required amount to the melt. At Karlsruhe Institute of Technology (KIT) a facility was erected for the production of lithium orthosilicate breeder pebbles. In this facility, e.g. lithium hydroxide and silica powders are molten in a crucible at temperatures above 1300 °C, and pebbles are then produced by the generation of single droplets through a nozzle (Fig. 1). Breeder pebbles with a narrow size distribution and a mean diameter of 700 µm are obtained. The development of the process aims at a controlled droplet generation to be monitored by a high-speed camera and at optimizing the cooling path to control the crystallization of the droplets.

Like the solid breeder blanket concept of Japan, the European blanket concept features the breeder ceramics as pebble beds inside a steel structure. The pebble bed will not take part in the structural rigidity of this structure, but the pebbles need to withstand forces originating from thermal expansion mismatches and neutron irradiation. Therefore, rigid and relatively defect free pebbles have to be fabricated.

Presently mixed ceramics containing lithium orthosilicate as the main phase and lithium metatitanate as a secondary phase are produced by the melt-based process with the aim to combine the advantages of both favored tritium breeder materials, i.e. the high intrinsic rigidity of lithium metatitanate and the high lithium density of lithium orthosilicate. Recent results showed that both phases, lithium orthosilicate and metatitanate, crystallize dendritically and form a very fine grained microstructure that is supposed to be very stable even after longer annealing times (Fig. 2). It was also proved that additions of lithium metatitanate can indeed enhance the mechanical properties of lithium orthosilicate pebbles.

Fig. 1: Pebble fabrication by the melt-based process in an experimental facility (left). Three nozzles are operated simultaneously in this case. Each melt jet consists of individual droplets. The solidified ceramic pebbles exhibit a mean diameter of 700 µm (right).

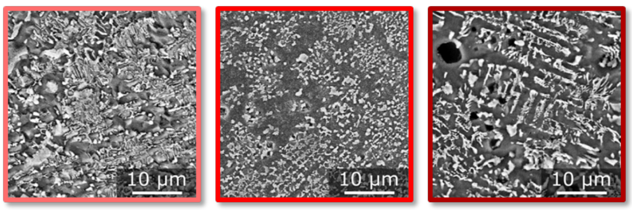

Fig. 2: Microstructure of mixed breeder pebbles containing lithium orthosilicate (dark grey) and 20 mol% lithium metatitanate (light grey) in the initial state after production (left), and after annealing at 950 °C for 1 week (middle) and 3 weeks (right).

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DEMO R&D Activity

R&D activities in JA Task 4 (Advanced Neutron Multiplier)

Beryllium intermetallic compounds (beryllides) have become promising materials for advanced neutron multipliers. Establishing the fabrication technique for beryllide is a key issue. In JAEA, novel granulation process of beryllide using plasma sintering and rotating electrode methods has been developed.

I. Beryllide synthesis

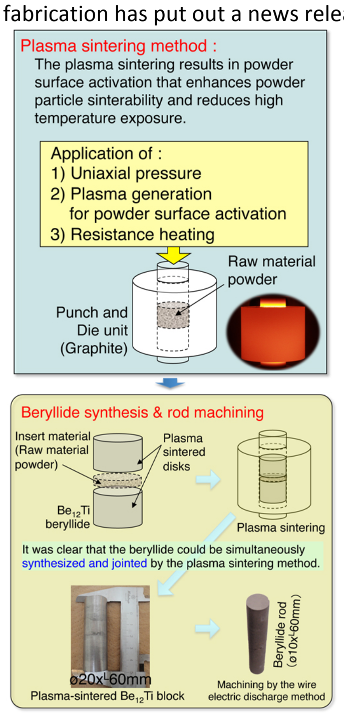
Beryllium intermetallic compounds (beryllides) such as $Be_{12}Ti$ are one of the most promising advanced neutron multipliers. In order to fabricate the beryllide pebbles, beryllide with shapes of block and/or rod is necessary when a melting granulation process is applied such as a rotating electrode method. However, beryllide is too brittle for the fabrication of blocks or rods by these methods. As conventional methods, a hot isostatic pressing (HIP) method and a casting method have been proposed for beryllide synthesis. It was clarified that these methods had some problems such as its complicated process, time-consuming characteristics and difficulty in composition control.

A plasma sintering method has been proposed as new technique which uses a non conventional consolidation process, because this method is simple, and is easy to control (see Fig.1). It was clarified that the beryllide could be simultaneously synthesized and jointed by the plasma sintering method in the insert material region between two beryllide blocks, with no variation of the phase and hardness. Beryllide block of $Be_{12}Ti$ with 20 mm in diameter and 60 mm in length has been successfully fabricated by the plasma sintering method. From the trial examination result of machinability of beryllide, beryllide rod with 10 mm in diameter and 60 mm in length could be machined by a wire electric discharge (WED) method from the plasma-sintered beryllide block. It has become clear that difficult-to-machine beryllide could be machined by the WED method accurately and efficiently.

II. Beryllide pebble fabrication

Using this plasma-sintered beryllide rod machined by the WED method, prototype pebble of beryllide was performed by a rotating electrode method as one of the melting methods (see Fig.2). The rotating electrode method is a method for producing metal powders where the end of a metal rod is melted while it is rotated about its longitudinal axis. Molten metal is centrifugally ejected and forms droplets that solidify to spherical powder particles. From the result of trial fabrication examination, the prototype pebbles of $Be_{12}Ti$ with 1 mm in average diameter were successfully fabricated.

This achievement of success of prototype pebble fabrication has put out a news release on 22nd June 2012.



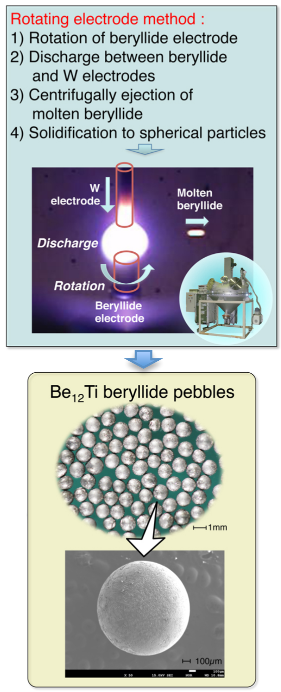


Fig. 1 Beryllide synthesis *Fig. 2 Beryllide pebble fabrication*

IFERC-N-2012-16 (*No.6, 31 August, 2012*)

DEMO Design Activity

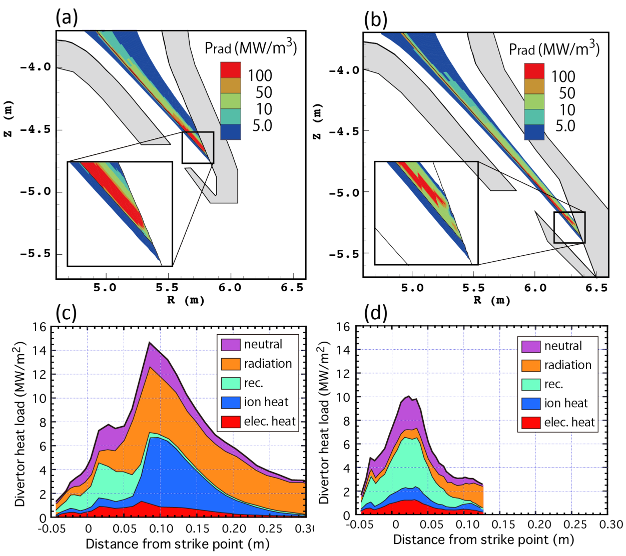
Progress of power handling design toward DEMO

Handling of a huge exhausted power from the core plasma is one of the most crucial issues for fusion tokamak reactor design. For example, in the conceptual DEMO reactor, SlimCS, the exhausted power into the scrape off layer and divertor region is 500-600 MW. On the other hand, the desirable heat load on the divertor target is expected to be less than that of ITER (< 10 MW/m^2).

The power handling design and related physics studies are being progressed by using a suite of integrated divertor codes SONIC. The basic divertor design concept is similar to that of ITER. The SONIC simulation has demonstrated that the detached divertor plasma was formed by installing the V-shaped corner and using the gas puff and Ar impurity seeding. Although the heat load due to the plasma transport was significantly reduced by the above effects, the contribution of the impurity radiation and the surface recombination to the target heat load became large. As a result, the total heat load is still much higher than the desirable level, as shown in Fig. (c).

For further reduction of the total heat load including contributions of the impurity radiation and the surface recombination, optimization of the divertor geometry, such as length of the divertor leg, target inclination, location of exhaust slots, etc, is one of possible solutions.

For example, the effect of the divertor leg length (length from divertor target to X-point) on the power handling is shown in the figures. The divertor plasma temperature decreases due to the longer magnetic field line and deep V-shaped corner. In addition, peak of the Ar impurity radiation, which is located close to the outer divertor target in the standard divertor geometry, moves upstream (Figs. (a) and (b)). The former decreases the heat load to the outer divertor due to the ion conduction and convection, and the latter reduces the heat load due to the impurity radiation. The peak of the total heat load on the outer divertor decreases by about 30 % (Figs. (c) and (d)).



Figs. Spatial profiles of the impurity radiation power (a)(b) and the divertor heat load along the outer divertor (c)(d). The standard case (a)(c) and the long-leg case (b)(d)

IFERC-N-2012-17 (*No.6, 31 August, 2012*)

Meetings

Preparatory Working Group on REC has face-to-face meeting in Barcelona and Cadarache

The Preparatory Working Group (PWG) on the ITER Remote Experimentation Centre (REC) had a face-to-face meeting at Barcelona and Cadarache in June to develop an overall plan for the REC.

During the first half of the meeting in Barcelona, the discussion was focused on the schedule, functions and technical specifications including technical options. The second half of the meeting in Cadarache addressed the technical discussions with participants from ITER Organization (IO) and Tore Supra, where 6 experts from IO and 3 from Tore Supra joined the discussions. Very valuable information was exchanged among the participants, including the development schedule of the ITER CODAC, which is closely related to that of the REC. Experiences of remote experiments and related technology were also presented from JT-60 and Tore Supra, including a remote experiment demonstrated for JT-60 between JAEA Naka and IPP Garching in 2007 (see the article in JT-60SA newsletter). All the participants welcomed their collaboration on the REC. The meeting ended successfully with essential agreement on the main points, as a basis for the overall plan.

During the meeting in Cadarache, the PWG members visited the ITER construction site. They were very impressed at its progress and scale, which expanded their view for future remote participation in experiments from Rokkasho in Japan.