PROGRESS OF ITER R&D IN THE JAPANESE HOME TEAM

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ABSTRACT

In this paper we report on the progress of ITER R&D in the Japanese Home Team. In close collaboration with more than 30 top level industries and many universities, various R&D tasks assigned to the Japanese Home Team have been conducted. In particular, reactor structures such as superconducting coils and vacuum vessel, in-vessel components such as blanket and divertor, out-vessel components such as heating and current drive systems, remote maintenance technology, and seismic isolation systems have shown significant progress.

INTRODUCTION

Fusion energy can be obtained by the high temperature, high density plasma confined in a strong magnetic field. The fusion reactor is operated in sustained burn, where produced fusion power is far higher than those necessary for heating and plasma current driving. Though kinetic energies of ion and electron constituting the plasma are extremely high (>10 keV), densities of those particles are almost million times smaller than normal gasses of the atmospheric pressure.

In the case of D-T burning, a collision of such high energy D,T ions produces fusion energy of 17.5 MeV, where 14.1 MeV is shared by the produced neutrons and the rest is shared by the produced He ions (Figure 1). The neutrons travel straight forward and their energy can be captured by the blanket materials and transferred to the coolant. The He ions are trapped by the magnetic field in the core plasma, and dispatch their energy to the plasma through coulomb collisional relaxation. Thus, out of the total fusion power, 80% is distributed to the blanket through neutrons and 20% to the plasma through He ions. In the steady state, power loss from the plasma through radiation and conduction can be balanced from the power input by the internal fusion He ions and by external heating systems. While the radiation loss power is distributed over the blanket first wall, the conduction loss power are conveyed by the low temperature plasma outside of the so-called "separatrix" toward the divertor region. At the divertor region, most of the power is expected to be dissipated due to collision with neutral gasses , and the residual small fraction is transported by plasma particles to the solid divertor plate, and then to the coolant.

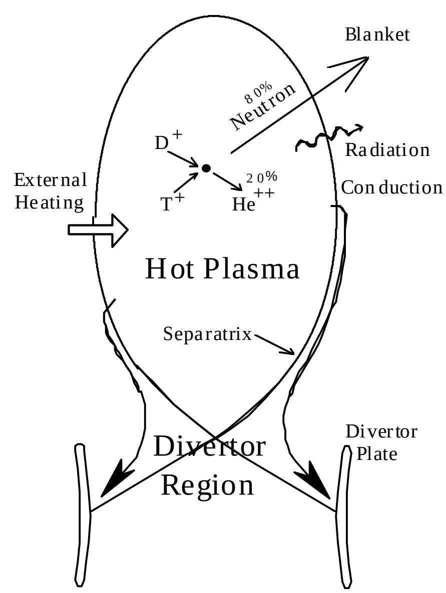


Figure 1 Heat Flow Schematic of Reactor

To realize those schemes of the fusion reactor, the ITER (IAEA/ITER EDA/DS/07, 1996) is the first device that comprises almost all the components necessary for the future commercial reactor. The only one exception is the lack of the breeding blanket for electricity generation. The important hardware components are categorized:

1) Reactor Structural Components:
   Superconducting Coil System and Vacuum Vessel
2) In Vessel Components:
   Blanket, Divertor and Limiter
3) Out of Vessel Component
   Heating and Current Drive System, Diagnostics System, Tritium Plant System, Refrigeration System, Vacuum Pumping System
4) Remote Maintenance System
   In Vessel Maintenance, Out-of Vessel Transporter
5) Service Plant
   Building, Power Supply System, Cooling System, Other Utilities

In order to ensure the feasibility of the ITER design, it is necessary to conduct technology R&D including development, manufacturing and testing of scalable models. Large scale R&D programs are developed after the outline of the ITER design is fixed where multi-parties jointly carry out the work. Although these projects took a few years to build up, now they are under integrated testing or under final manufacturing or assembly.

In the Japanese Home Team, the role of JAERI is to coordinate and develop a plan for the assigned R&D in detail, to procure components, assemble, and carry out testing. For ITER, deeper and wider involvement of industries is needed because of increasing role of manufacturing and fabrication technology. Therefore, some essential portion of the R&D is carried out at works of these industries. In the following, we present some typical examples of the R&D underway.

REACTOR STRUCTURE

Superconducting Coil

Development of superconducting coil for ITER is the most important because they occupy a significant part of cost, and their performance is critical to the expected achievement of plasma confinement, duration and control.In particular, the Central Solenoid (CS) Model Coil aims at verifying, with a reduced scale, the fundamental performance and manufacturing feasibility of the ITER CS coil at the same level of conductor current of 46 kA, generating a magnetic field of 13 T, and the rate of field change of1T/sec. The Model Coil consists of 10 layered inner module, 8 layered outer module and a single layer insert coil. The JA Home Team is responsible for fabrication of the outer module and the single layer insert coil made of the superconducting cables developed and fabricated jointly by the collaboration of US, EU and JA home teams.

Technological significance of the conductor required for ITER is far advanced as compared with already developed conductors: that are up to 8T at medium current levels, or up to 12T but at a low current level (Figure 2). Only feasible candidate that will satisfy ITER requirements by the time of construction is the use of Nb_3Sn for superconducting material. However, this material requires careful handing in avoidance of excess stress in contrast with the established stable material of NbTi.

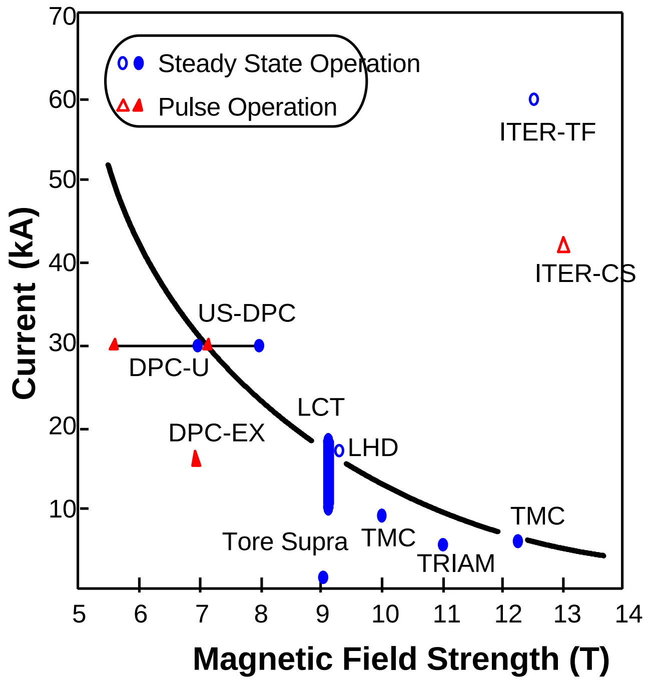
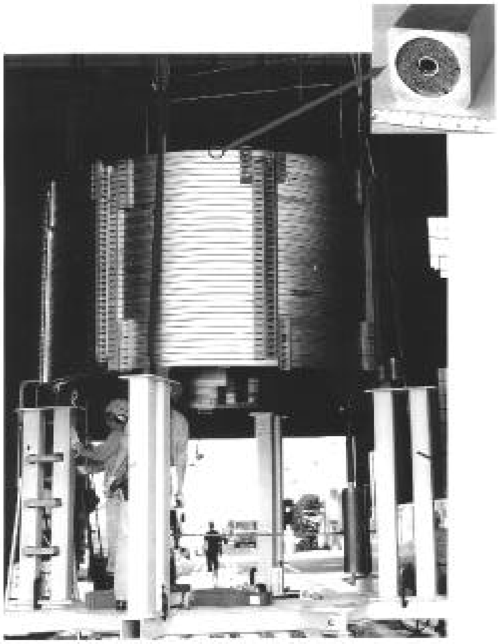


Figure 2 Characteristics of the superconducting conductors for ITER compared with those for other devices or coils.

High critical current and low pulse losses under the pulse operating condition are required for the ITER superconducting strand; the basic element of the superconducting conductor. This ITER requirement was satisfied by the development of a low heat generation strand whose performance is one fifth of those developed before. Industrial production of such strands started after thorough checkout of the production quality and the planned 13.6 tons have been produced. About 200 m long, 1000 strands are Cr coated and twisted in a form of a cable to minimize loop coupling with pulsed magnetic field. Thus produced cables are transported to EU for pull through into the INCOLOY (high nickel based steel) tube and compacted to form a conductor. These conductors are transported to US and Japan for inner and outer module fabrication, respectively; the most difficult process. A pair of conductors, each 5 cm x 5 cm square cross section, is then wound to form a layer. Since eight layers must be stacked tightly in one module without interference between any layers, and without loss of mechanical integrity, each layer must be wound very precisely. After winding, these layers are heat treated (kept for 240 hours at 650 ^{o}C) to form a Nb_3Sn superconducting alloy in a big vacuum furnace. In this process, special caution must be made to keep the oxygen content far below o.1 ppm to avoid SAGBO (Stress Assisted Grain Boundary Oxidation) crack in the INCOLOY couduit surface. A large scale international investigation of this issue was conducted cooperatively before the real heat treatment on model coil conductor. As a result both US and Japan could be successful in the heat treatment avoiding SAGBO in the summer of 1997 (Figure 3). These windings are then tape insulated and stacked from the inner to the outer layer, and epoxy insulator impregnated.

The test facility for the CS model coil was complete in 1996, and is ready for testing. The main feature of this system is its world biggest capability of 50 kA current at 4.5 kV using JT-60 pulsed power supply. The rated refrigerator capacity is 5 kW at 4.5 K and the vacuum chamber for the model coil is 6.5 m in diameter and 9.5 m high.



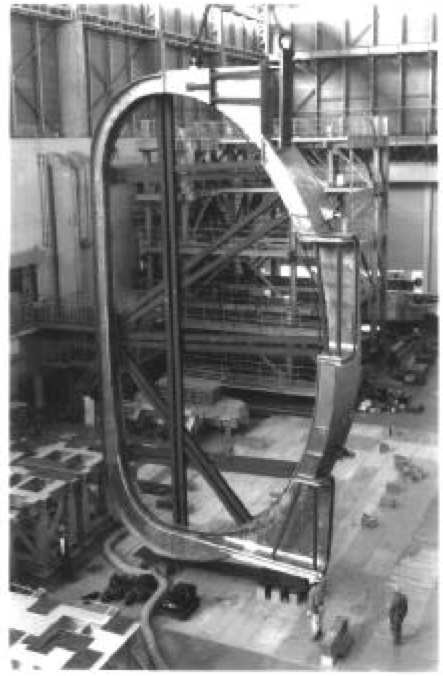


Figure 3 First layer after the heat treatment for the CS model coil. The conductor is made of twisted 1000 Nb_3Sn strands inserted and compacted in a high nickel steel (Incoloy) jacket

Figure 4 Vacuum vessel full-scale sector model before in-situ welding along the poloidal contour.

Vacuum Vessel Sector

ITER Vacuum Vessel constitutes the first vacuum boundary for the plasma, and contains in vessel components such as blanket and the divertor cassette. It takes a role of nuclear shield protecting the superconducting magnets and also removes small nuclear decay heat of about 3 MW. It is composed of D-shaped stainless steel double walls with connecting ribs between them. Its structure is to keep high electrical resistance along the torus while keeping mechanical integrity against transient electromagnetic forces (15,000 ton) and external atmospheric pressure. Its dimension is about 15 m high and 9 m wide with the wall thickness of 4-6 cm. The biggest uncertainty for this design is the manufacturing feasibility of attaining high accuracy (maximum allowable tolerance of $\pm$ 20 mm) for this kind of huge structure, since the vacuum vessel is the geometrical reference to in-vessel components defining the position of the plasma boundary. The other concern is whether the vacuum vessel can be remotely cut and welded again should the worst trouble take place. For these purposes, a sector model R&D is planned and implemented. It is equivalent to one of the 20 modules of the ITER Vacuum Vessel, and is composed of two half sectors with full-scale cross section. Each half sector was made by the different industry with a different fabrication and assembling approach (TIG, MIG and electron beam welding). The outcome of both sectors has shown sufficient tolerance of $\pm$ 3 mm and could foresee to satisfy the level of the target after in-situ welding of the poloidal contact surface between the two half sectors (Figure 4).

IN-VESSEL COMPONENT

Blanket

Blanket is one of the most important elements of the fusion reactor, where the 14 MeV neutrons are trapped and thermalized. The energy transported by neutrons is volumetrically cooled and transported by the coolant, but its first wall surface is illuminated by the radiation from the plasma as well. Thus the most sensible portion of the blanket design is its first wall. Stainless steel is selected in ITER for the base structural material of the blanket, because there are plenty of database to justify the use during lifetime of ITER (< 1 MWa/m^2), but it is not intended to be used for the future reactor blanket. To avoid excessive temperature increase and thermal stress, the first wall should have high cooling capability, and for this reason high conducting DS copper plates and dense array of cooling tubes are used in this region. The blanket module is also subject to large electromagnetic forces. To satisfy these requirements, it is important to develop fabrication technologies of a blanket module, and most importantly stiff bonding of first wall to the main body of the blanket durable against lifetime heat cycles. A reliable metal joint (DS Cu - SS) surface can now be produced after a series of development from the small scale to the medium scale models. Figure 5 shows 1/2-scaled module successfully fabricated by one-step Hot Isostatic Pressing (HIP) bonding technique.

Divertor

Divertor is the most stringent component to handle highest heat flux to the material wall. Heat flow across the separatrix magnetic flux surface is conveyed by the low temperature plasma outside the separatrix. Major part of the heat flux is expected to be dissipated at the divertor region through ionization and excitation of neutral fuel gasses or of impurity gasses, yet the residual heat flux conveyed by the plasma is demanding. Against variety of operation as experimental reactor, in particular, before optimization of the operational scenario, surface heat flux requirements to the ITER divertor plate are set to be 5 MW/m^2 steady state, and 20 MW/m^2 for short pulses of 10 sec. Number of small sample tests and improvements have been carried out since relatively early stage of the EDA, and basic geometry has been successfully developed that can withstand these heat fluxes for specified heat cycles. The cooling tube is made of copper alloy screw tube, which pierce an array of the segmented blocks made of CFC (carbon fiber reinforced composite) armor plate on top of the copper alloy block for heat sink. In addition, a full length 1.3 m long curved element was made to confirm manufacturing feasibility and for irradiation test in the ion beam test bed (Figure 6). On these basis, a full-size vertical target plate is fabricated, a half of its surface is covered by the tungsten armors and the other half is covered by the CFC armors. It will be tested soon, and the similar ones will be shipped to the US for the integrated testing as the Divertor Cassette R&D.

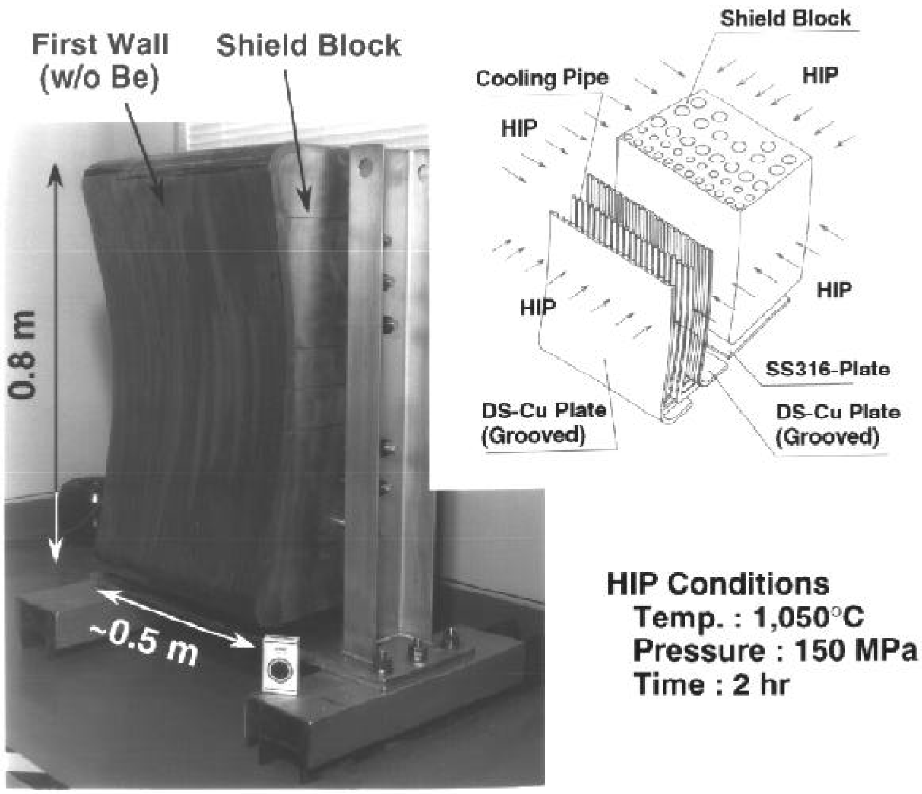


Figure 5 A half-scale module successfully fabricated by one-step Hot Isostatic Pressing bonding technique.

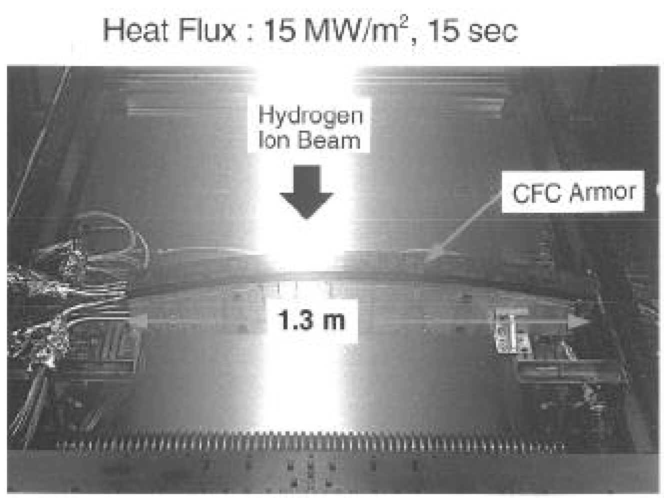


Figure 6 Full length 1.3 m long divertor element under ion beam irradiation test.

OUT-VESSEL COMPONENTS

In the ITER out-vessel components, developments of heating and current drive systems require most advanced technology. Among several methods, efforts are focused in JA Home Team on the Neutral Beam and on the ECRF system, because they have high applicability to the future reactors, and because there have been cumulative technology basis on these systems by the JT-60 program.

Neutral Beam

In order to inject significant power into the torus for heating the plasma and driving plasma current, energetic neutral particle beam has been used. The neutral particle beam, without being affected by the magnetic field is, injected straight forward into the torus, ionized and trapped in the plasma very efficiently. Such neutral beam can be obtained by the conversion of the normal positive ion beam, but with the increase of the plasma line density, the neutral beam energy must be high enough so that the only practical way is to use negative ion beam to achieve high conversion efficiency. Thus the major issues is to develop a high current negative ion source, and a high energy accelerator. Target design values for the ITER ion source are beam energy of 1 MeV, beam current of 40 A (20 mA/cm^2). The dual path R&D approach have been adopted. Namely one is to develop high energy acceleration, and the other is to develop high current scalable to the ITER ion source. To date, toward 1 MeV acceleration, negative ion beams with good beam optics have been accelerated up to 975 keV using a multi-aperture multi-gap electrostatic accelerator. For large current extraction, the intermediate scale source has produced 13.5 A at 400 keV negative deuterium beam and this source has been used to the JT-60 experiment for current drive purpose. These results show when compared with other project plans and industrial applications, that fusion R&D leads such advanced technology (Figure 7).

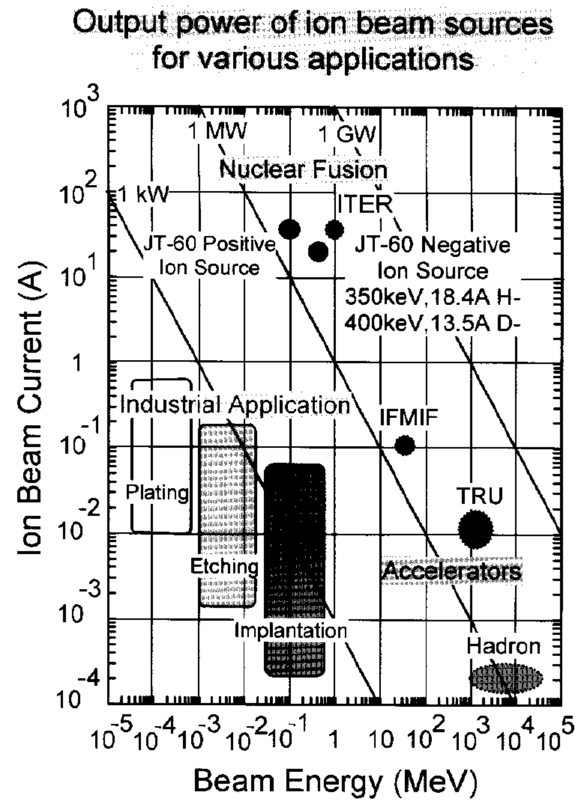


Figure 7 Output power of ion beam for fusion and various applications and requirement. Ion beam for fusion is leading multi-megawattsource development.

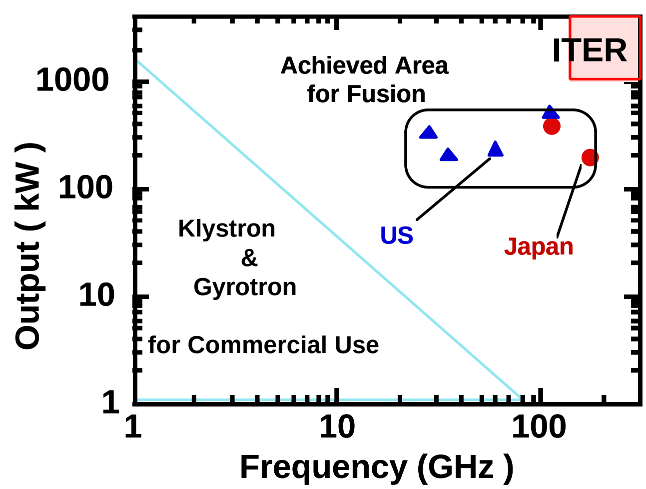


Figure 8 A 170 GHz Gyrotron under development for heating and current drive aims by far the commercially available performance.

ECRF

For the initial plasma production and heating and current drive purposes, ECRF systems have been extensively developed. Toward the goal of ITER, 1 MW Gyrotron oscillator of continuous rating is required at high frequency of 170 GHz and with a power conversion efficiency exceeding 50 %. The RF power is obtained by the conversion of electron beam power in the Gyrotron, but so far the level of conversion efficiency had been less than 35 %.

High efficiency for ITER can only be achieved by an electrostatic energy recovery concept developed at JAERI, where spent beam energy can be recovered by electrostatic deceleration. The concept has already been demonstrated at somewhat lower frequency (110 GHz, 400 kW, 4 sec), and is being applied to the 170 GHz tube. Overall the JA Home Team has presently achieved 500 kW, 0.7 sec at 170 GHz and is now improving oscillator by optimization of geometry and suppression of power loss at the windows through the introduction of CVD diamond. Again, these developments are far ahead of the presently available high frequency oscillators (Figure 8).

REMOTE MAINTENANCE

Blanket Maintenance

Remote maintenance systems for in-vessel components are of particular importance for ITER stable and efficient operation. The JA Home Team is involved in the blanket and the divertor remote maintenance systems and is the main implementation party for the blanket maintenance system. The blanket module typically made of 2 m wide 1 m long, 40-50 cm deep, 4-5 tons must be replaced remotely by the vehicle type maintenance system. The system consists of an automatically deploying ring rail that is supported at four ports along the torus, a moving vehicle on the rail, the manipulator mounted on the vehicle and the end effector that grasps the blanket module. The partial models were fabricated and various tests have been carried out successfully (Figure. 9). On this basis, a full-scale remote handling equipment has been fabricated and is under final assembly at JAERI.

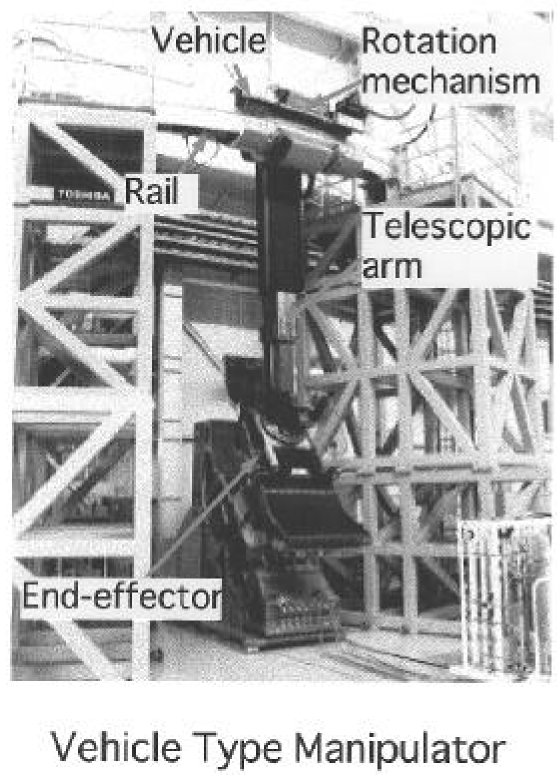


Figure 9 Blanket module remote handing R&D. The prototype scale R&D is underway for vehicle, manipulator and rail transporter.

Pipe Welding and Cutting Tool

Remote maintenance of in-vessel components such as blanket and divetor requires also cutting and welding of the cooling tubes remotely. For this purpose pipe welding &cutting tool has been developed and demonstrated. The internal access tool cantravel through 10 cm diameter pipe with bent radius down to 40 cm. The travel speed is 50 cm /min, and can cut/weld the manifold and the branch pipes up to 6 mm thickness by YAG laser of 2-4 kW.

SEISMIC DESIGN

In the ITER reference design, the Tokamak is designed to withstand the IAEA SL-2 class seismic acceleration of 0.2g. When the actual site is selected, where higher seismicity is required, there are two choices. One is to redesign the support structure of the Tokamak, but this is not a simple job, because mutual support between the magnet system (which is cooled at liquid helium temperature) and the vacuum vessel (which is operated at room to 200 ^{o}C) has to be minimized to suppress heat load to the cryogenic system and the support should be flexible to allow mutual displacement due to temperature change between operation and assembling at room temperature. The other is to introduce seismically isolated structure in the basement of the building or in a part of the building. Figure 10 shows schematically such a concept applied between the pit of the building and the base of the Tokamak building. The Tokamak and its peripheral components are installed in the pit structure whose base wall is supported by the multilayered elastomeric isolators. Such a system has recently been applied to some of the new buildings to relax acceleration, and its effectiveness was demonstrated by the Hanshin earthquake in 1996. For ITER application, requirement for the isolator may be the diameter of 1.2-1.5 m vertical stress of 100-150 kg/cm^2 and the oscillation period of 3-4 sec. as compared with the already developed isolators for public use of 80 kg/cm^2. Thus it may be well within our reach to develop ITER use, and for this purpose R&D has been started.

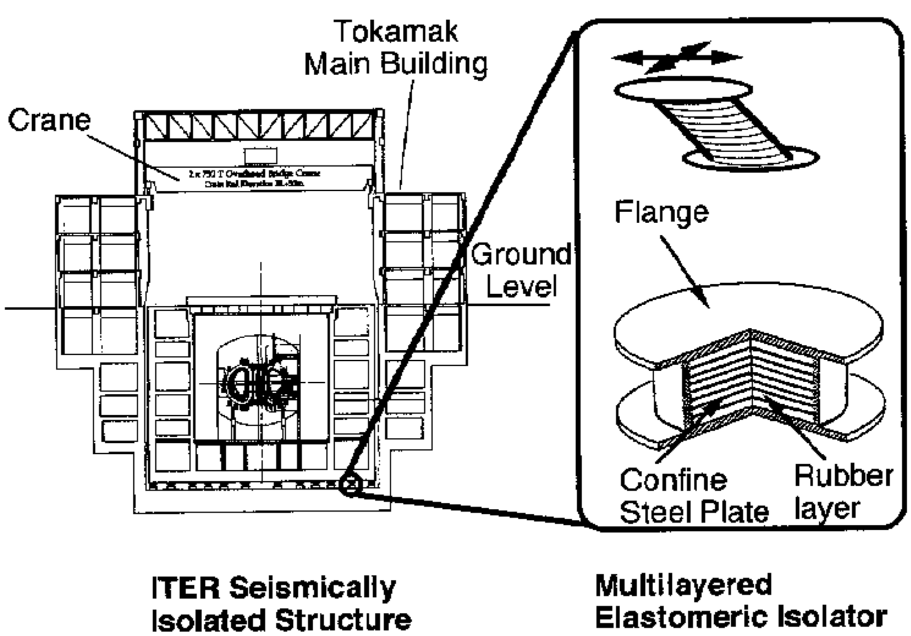


Figure 10 Seismic design by supporting Tokamak pit base using multilayered elastomeric isolators.

SUMMARY

Extensive R&D is underway by the Japanese Home Team to validate integrated engineering design of ITER. In particular, there has been a significant progress during the latter half of the EDA both on the component R&D and on the large scale prototypical R&D. All the data so far obtained are supportive of the design, and together with the expected additional data in near term they will permit us to proceed into ITER construction phase with sufficient technical confidence.

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