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Overview on materials R&D activities in Japan towards ITER construction and operation

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Abstract

This paper presents an overview of ITER-supporting materials R&D activities and major achievements in Japan during the period from the Co-ordinated Technical Activities to date. In view of the completed engineering design of ITER during the Engineering Design Activities period, R&D efforts since then have been focused on: those for reduction of component fabrication cost; those in support of domestic preparations of a structural technical code for construction; and those necessary for operation, and been extended to component-level testing rather than pure material testing. They cover materials R&D for in-vessel components, vacuum vessel, cryogenic steels of superconducting magnets and diagnostics components. Major achievements in each R&D area are highlighted and their impact or implication to the design, construction and operation of ITER is presented.
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1. Introduction

Extensive R&D had been conducted in Japan during the ITER Engineering Design Activities (EDA) mainly for selecting design choices and completing the engineering design. During the subsequent phases, in view of the progress of international negotiations towards construction and operation, R&D in this area has been extended to: those for reduction of component fabrication cost; those in support of domestic preparations of a structural technical code for construction; and those necessary for operation of the machine.

From the viewpoint of reduction of component fabrication cost, cost-effective fabrication routes have been pursued for the divertor, and their performance under ITER relevant conditions has been examined. For the jacket material of the central solenoid conductor, alternate materials have been explored with a view to relaxing the constraints on the design and fabrication processes.

As the vacuum vessel forms an essential part of the physical barrier to contain radioactive materials, its

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mechanical integrity, in particular, T-welded joints with a partial penetration, has been examined in support of domestic preparations of a structural technical code for construction. Towards ITER operation, a new approach to remove co-deposited layers of tritium and carbon by means of laser irradiation has been explored, and neutron irradiation testing of key diagnostics components has continued.

In the following sections, outline and major achievements on ITER-supporting materials R&D activities in Japan after the EDA to date are described in a systemwise manner from internal to external components.

2. In-vessel components

2.1. Reduced-cost options for tungsten armor and copper cooling tube of the divertor

Two reduced cost options have been explored for application to the tungsten armor and copper cooling tube of the divertor: a bundle of tungsten rods hot-pressed onto a heat sink as an armor configuration ('rod-shaped tungsten armor' [1]); and a copper cooling tube with internal fins machined by simple mechanical threading ('screw tube' [2]). Both options have simple fabrication routes with reduced fabrication cost, and their performance has been examined under ITER heat load conditions.

The rod-shaped tungsten armor is a cluster of commercially available sintered tungsten rods, pressed into a bore of the copper heat sink, and then hot-pressed. It has another advantage of reduced thermal stress between the armor and the heat sink and increased flexibility in selecting the thickness of the armor. A mock-up, shown in Fig. 1, was fabricated and electron beam irradiated under a heat flux of 10 $MW/m^2 \times 15$ s. It showed no failure over 3000 cycles, and applicability of this option has been demonstrated.

The screw tube is a copper cooling tube with internal fins in a helical and triangular shape machined by simple mechanical threading. By using a conventional threading process, cost reduction can be expected. Heat removal capabilities have already been confirmed, comparable with those of the twisted tape [3]. Mock-ups made of CuCrZr with M10 threading were fabricated and adequate thermal-fatigue performance was demonstrated under the ITER reference conditions of 20 MW/m^2 and 300 cycles. Accelerated tests were continued under 20 and 30 MW/m^2 till the fracture, which occurred at 4500th and 1400th cycles respectively. Fractgraphic observation showed fatigue cracks initiated from the outer surface of the tube and propagated toward the inner surface. Though stress concentration at, and crack initiation from the root of the internal fins were concerns for this concept, above results has indicated that the fins have no dominant effect on their fatigue lifetimes.

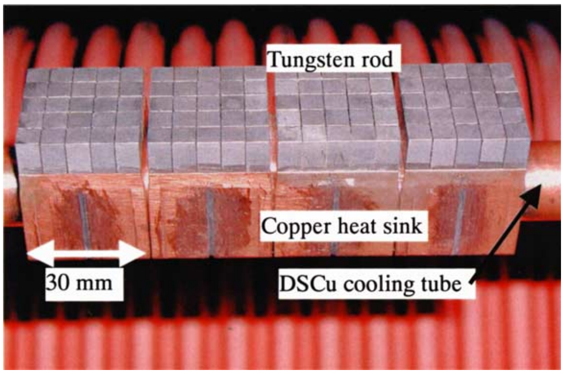

Fig. 1. A divertor mockup of 'rod-shaped tungsten armor' for thermal cycle testing. A bundle of commercially-available sintered tungsten rods hot-pressed onto the copper heat sink to reduce fabrication cost and relax thermal stress between the armor and heat sink.

2.2. Mechanical properties of Cu-alloys after heat treatment processes

Tensile and fatigue properties of Cu-alloys (DS-Cu and CuCrZr) have been obtained, taking into account realistic heat treatment processes during the fabrication route of the first wall; (1) HIP bonding of stainless steel (SS) and DS-Cu (1050 °C, 2 h) and then HIP bonding of DS-Cu and Be armor (620 °C, 2 h), and (2) HIP bonding of SS and CuCrZr (1050 °C, 2 h), solution annealing (980 °C, 0.5 h, with gas quenching) and finally HIP bonding of CuCrZr and Be (550 °C, 1 hr, with simultaneous aging).

Typical results are shown in Fig. 2. The properties of the heat-treated DSCu were nearly comparable with those of the as-received ones, though the elongation was degraded at elevated temperature. On the other hand, ultimate tensile and yield strengths of the heat-treated CuCrZr were significantly degraded, which is supposedly due to insufficient cooling speed for the solution annealing and high temperature for the aging. Improvements of these properties can be foreseen by increasing the gas quenching speed and reducing the HIP bonding temperature with Be down to 500 °C.

2.3. Removal of co-deposited layers by excimer laser irradiation

During ITER operation tritium will be retained inside the vacuum vessel by co-deposition of tritium and carbon, and it is crucial to establish a rapid and effective technique for tritium recovery. While discharge cleaning and baking are deemed candidate methods, an

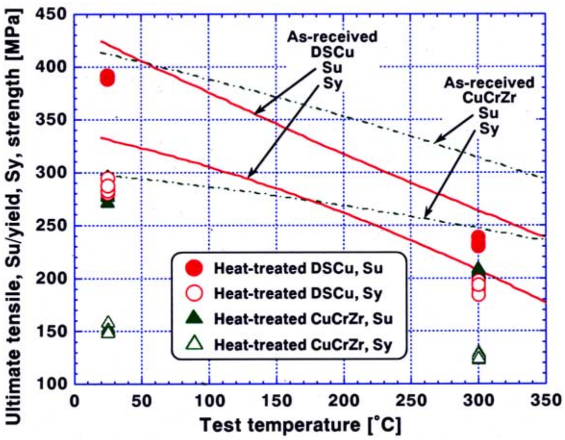

Fig. 2. Ultimate tensile and yield stresses of DS-Cu and Cu-CrZr after realistic heat treatment processes. Those of Ds-Cu are comparable to those of as-received materials, while improvements are needed for CuCrZr heat treatment processes.

alternative and promising technique based on excimer lasers irradiation has been explored [4]. This technique potentially has an advantage to recover tritium predominantly in the form of hydrogen molecules rather than tritiated water by breaking carbon–hydrogen.

A series of experiments have been conducted, by irradiating ArF or KrF excimer lasers onto the co-deposited layers formed in JT-60 or TFTR, to verify the feasibility, examine the removal mechanism and demonstrate the efficiency of this technique. Fig. 3 shows a typical result, an SEM image of the laser print of the JT-60 as-used carbon tile. As shown, the co-deposited layers with thickness of nearly 60 μm could be completely removed by 10 pulses' KrF irradiation at the fluence of 13.3 J/cm^2. Measurement of the surface temperature during irradiation and observation of the irradiated surfaces have brought the understanding that the mechanism to remove the co-deposited layers is full surface melting. Even though strong melting occurred during irradiation, the edges of the laser print were sharp and well-defined, and the substrate was not damaged. Based on these results, it could be estimated that the co-deposited layers of 100 m^2 and 50 μm thick can be removed within one week using 180 W KrF laser, if it is applied to ITER.

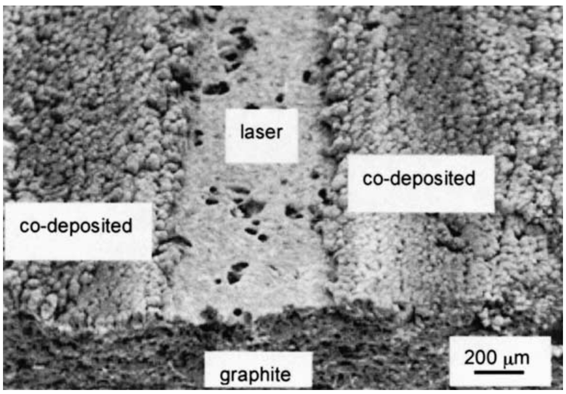

Fig. 3. An SEM image (45° view) of the laser print of the JT-60 co-deposited layers irradiated by 10 pulses of KrF excimer laser at the fluence of 13.3 J/cm^2. The co-deposited layers have been removed without any damage on the substrate graphite.

The released gases during the repetitive laser pulses were measured, and it was confirmed that hydrogen (tritium) was released predominantly in the form of hydrogen molecules as expected. Based on these results, the excimer laser ablation has been proposed as a very promising technique for tritium recovery from the co-deposited layers in ITER.

3. Vacuum vessel

The vacuum vessel forms a main part of physical barriers against release of radioactive materials, and its mechanical integrity needs to be fully assured including welding joints. T-welded joints with a partial penetration are to be applied for the vacuum vessel, and the strength of this type of welded joints has been examined. Large-sized, T-welded SS316L joints were prepared by applying candidate welding processes and joint configurations, and tensile and tensile fatigue tests were conducted [5]. The tensile tests showed adequate joint efficiencies over 100% except for the through-wall electron beam (EB) welded joints (penetrated area of around 30%) whose joint efficiency was 93%. While the definition of the joint efficiency needs careful considerations, these data are quite promising.

Tensile fatigue tests were conducted for three types of T-welded joints prepared by: full penetration EB (for a comparison purpose); partial penetration through-wall EB; and partial penetration metal active gas welding (MAG). The fatigue strength reduction factors K_f obtained for each joint were 2, 5 and 6, respectively. As the K_f value of partial penetration joints with proper welding quality is expected to be less than 4–5, these results suggest the partial penetration MAG-welded joints are not acceptable. The partial penetration through-wall EB-welded joints are marginal. However, taking into account the conservative test conditions, they are assessed as acceptable.

In order to evaluate neutron irradiation effects on the mechanical properties of the welded joints, tensile tests were conducted for the TIG, EB and MAG welded joints irradiated at JMTR up to 0.5 dpa at 200 °C [6]. While all of the data showed radiation hardening, they still retained sufficiently large total elongation and reduction of area except for the MAG-welded joints. A number of inclusions were observed on the fractured surfaces of the MAG-joints. Through these studies, it could be concluded that the partial penetration T-welded joints are acceptable and the irradiation effects are negligibly small. It is also recommended to avoid MAG-welded joints unless its welding quality is improved.

4. Cryogenic steels

Ni-based Incoloy 908 was used as a jacket material of the central solenoid model coil (CSMC) conductor during the EDA due to its low thermal expansion coefficient and adequate mechanical properties, and the CSMC successfully demonstrated the required superconducting performances. However, because this material has a disadvantage of stress accelerated grain boundary oxidation during the Nb_3Sn heat treatment (240 h at 923 K) process [7], efforts have been continued after the EDA to explore an alternate jacket material.

Improvement of Nb_3Sn strand performances during the EDA has allowed use of stainless steels as a jacket material. A high manganese stainless steel JK2 developed in Japan during the EDA satisfied the requirements of both thermal expansion and mechanical properties at solution treatment conditions [8], but improvement of the mechanical properties after aging was a remaining issue for JK2. To this end, a modified grade JK2LB (0.03C–22Mn–13Cr–9Ni–1Mo–0.24N–0.003B), with reduced content of carbon and addition of boron, has been pursued with a target of reducing precipitations at grain boundaries and giving adequate ductility and toughness even after aging [9].

Test samples were prepared from an intermediate billet and the final jacket of JK2LB produced by a mass production line, and tensile properties, fracture toughness, and crack propagation rate were measured at 4 K. The elongation at 4 K after aging was over 30% as shown in Fig. 4, drastically improved compared with JK2. The fracture toughness was also improved from 66 MPa$\sqrt{m}$ (JK2) to 93 MPa$\sqrt{m}$ (JK2LB, final jacket). The mechanical properties of weld metal of JK2LB were also examined at 4 K and all data including fatigue crack growth rates satisfied the ITER requirements. Based on these promising results, JK2LB has been selected as a reference material, instead of Incoloy 908, for the jacket material of the central solenoid.

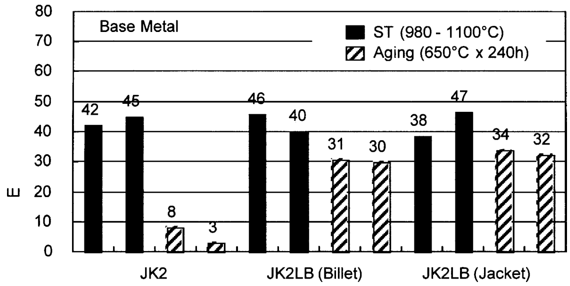

Fig. 4. Elongation of JK2 forged block, JK2LB billet, and JK2LB jacket at the temperature of 4 K after ST (solution treatment, 980–1100 °C) and after ST and aging process (650 °C, 240 h). Significant improvements can be seen for JK2LB after aging.

5. Diagnostics components

Irradiation tests on diagnostic components have been continued after EDA on key diagnostics components such as bolometer, optical fibers and magnetic coils.

Irradiation tests of a mica substrate bolometer were carried out at JMTR up to the fast neutron fluence of 6×10^{23} n/m^2. Significant increase of the resistance of the meander (zigzag-shaped resister laminated on the mica substrate) from 275 to 446 Ω was observed as shown in Fig. 5. Post irradiation examination confirmed that the gold meander contained 46% mercury, which was produced by the nuclear transmutation reaction. Breaks of the gold meander were observed in the post irradiation examination, and the use of gold meanders should be avoided [10].

Development and evaluation of radiation-resistant optical fibers have been continued with a view to installing them as close to the plasma as possible. International round robin experiments have yielded promising results and some optical fibers, such as Russia-made hydrogen loaded KU-1, could be used at the location close to the plasma for visible application with a limited life. Fluorine doped fibers showed a good radiation resistance in visible regions but recent reactor irradiation tests revealed that they had higher sensitivity

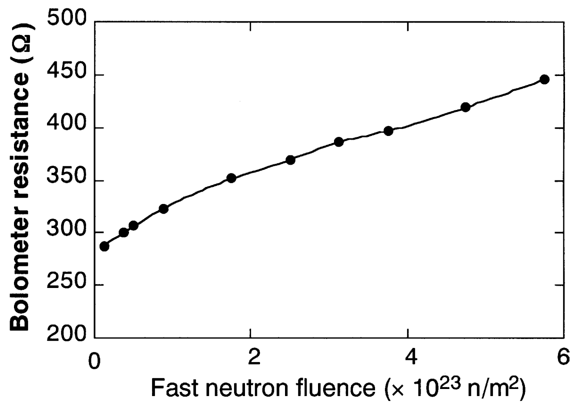

Fig. 5. Test results of a mica substrate bolometer irradiated at JMTR. Significant increase of the meander resistence was observed, which was caused by nuclear transmutation reaction of gold meander into mercury.

to the micro bending loss [11,12]. For infrared applications, several optical fibers could be found applicable with a life far beyond the ITER operation period.

Very low parasitic voltages in the order of mV may cause an error of the plasma position identification; R&D has been continued on radiation-induced effects on the magnetic coils. In situ measurements of differential voltages between the ends of two magnetic coils wound with MI cable have been carried out at JMTR. It was confirmed the radiation-induced electromotive force is not a serious issue for the magnetic coils, however, another parasitic voltage on the order of mV was observed, which could be explained by thermoelectric potentials induced by non-uniform transmutation of the copper core of the cable [13]. This effect should be seriously considered for the operations longer than 1000 s on ITER.

6. Conclusions

ITER-supporting materials R&D activities and major achievements in Japan during the period after the EDA to date are overviewed. R&D efforts during this period have been oriented to: those for reduction of component fabrication cost; those in support of domestic preparations of a structural technical code for construction; and those necessary for operation of the machine, and the following results have been obtained:

(1) Two options, 'rod-shaped tungsten armor' and 'screw tube', have been proposed for the divertor, mainly for reducing the fabrication cost, and their adequate thermo-mechanical performance has been demonstrated under ITER heat load conditions.
(2) An excimer laser irradiation technique has been proposed as a rapid and effective means to remove co-deposited layers of tritium and carbon retained inside the vacuum vessel, and effectiveness of this method has been demonstrated by applying to the carbon tiles with co-deposited layers as used in JT-60 and TFTR.
(3) As a course of domestic preparations of a structural technical code for construction, a series of mechanical tests have been performed for the T-welded joints with a partial penetration to be applied in the vacuum vessel fabrication. It could be concluded that this type of welded joints are acceptable from the structural integrity viewpoints.
(4) A new cryogenic steel, grade JK2LB, has been successfully developed for the jacket material of the central solenoid conductor, and it has been selected as the reference material, instead of Incoloy 908, due to improved mechanical properties even after the aging process.
(5) Irradiation testing has been conducted for key diagnostics components (bolometer, optical fibers and magnetic coils), and critical issues to be examined further have been addressed.

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