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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.

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.

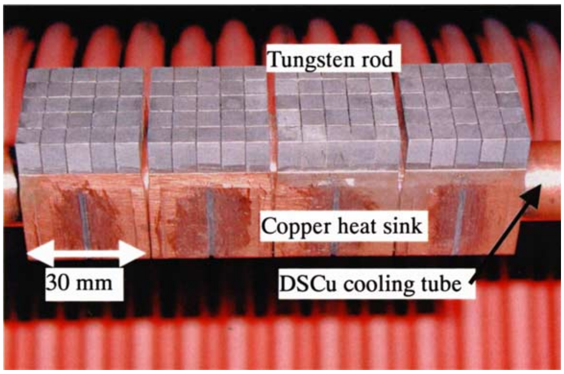

Fig. 1. A divertor mockup of 'rod-shaped tungsten armor' for thermal cycle testing. A bundle of commercially-available sintered tungsten rods hot-pressed onto the copper heat sink to reduce fabrication cost and relax thermal stress between the armor and heat sink.

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