§104. Neutron Irradiation Effects on Dissimilar Joints/Coatings of Blanket Structural Materials

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In a variety of fusion power plant concepts, such as, water-cooled lead-lithium (WCLL), helium-cooled ceramics/beryllium pebble bed (HCPB) and dual-coolant (DC) blanket systems, joining technologies of dissimilar materials are essentially required. Oxide dispersion strengthened (ODS) steel and tungsten (W) are considered as promising candidate materials for structural and plasma facing materials of the first wall and divertor components in fusion reactors. ODS steel shows excellent elevated temperature strength, corrosion resistance, radiation resistance, and W has high sputtering resistance and low tritium retention in fusion environment. Therefore, it is considered that the joining of ODS steels and W and its evaluation are a critical issue for the development of fusion application. However, the effect of neutron irradiation on dissimilar joints and coatings is not clear. The objective of this research is to investigate the effects of irradiation on the dissimilar joint as well as W itself of which the basic irradiation behavior is not fully understood.

Since the specimens for tensile tests after neutron irradiation are so-called miniaturized ones, we designed a new bending test-jig for the miniaturized bend specimens, as shown in Fig.1. The span length is 2.1 mm and the joint interface was located at the center of the specimen. The attachments were also produced to set the specimen at the adequate position to evaluate the joint strength. The impact properties of rolled pure W depended on the notch direction of the test specimens. The USE is higher in the specimens of which the notch is produced on the rolled surface of the W plate.

As for the neutron irradiation hardening of the vacuum plasma sprayed W, it was shown that the amount of hardening of VPS-W was 300 and 500 MPa after the irradiation at 773 K up to 1.6 and 2.7 dpa, respectively, in the HFIR in Oak Ridge National Laboratory. The formation behavior of the radiation damage structures is remarkably affected by the mobility of the point defects, such as vacancy which determines the diffusivity of the atoms in solid, and its clusters. It has been considered that the recovery stage of the vacancy clusters, that is, voids in W is in the temperature range above 1273K [1]. According to this expectation, the irradiation hardening of W will be observed up to around 1373K. Since neutron irradiation at very high temperature like 1273K is not easy, we conducted ion-irradiation experiment as a basic and supplementary study. Both the single and dual ion irradiation were conducted for pure W with iron ions and He ions. Fig. 2 shows the dependence of irradiation hardening on irradiation temperature after single iron irradiation up to 2 dpa. The irradiation induces hardening even at 1273 K. The hardening appears to show a peak at 973 K showing a small reduction with irradiation temperature. The microstructural examinations by TEM revealed that a number of dislocation loops were observed in the specimen irradiated at all the irradiation temperatures. No void was observed after the irradiation at 573 and 773 K. It can be concluded that the ion-irradiation hardening is due to the formation of dislocation loops.

[1] F. Ferroni, et al., Acta Materialia, 90(2015)380.

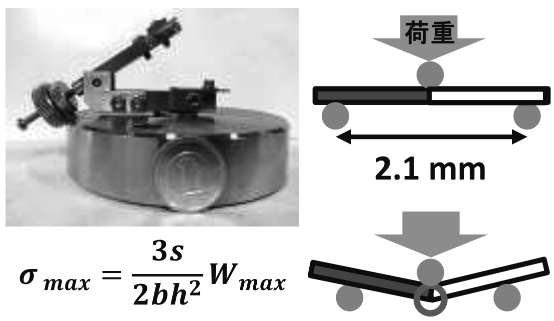

Fig. 1: The bending test jigs for miniaturized bending specimens irradiated in HFIR of ORNL.

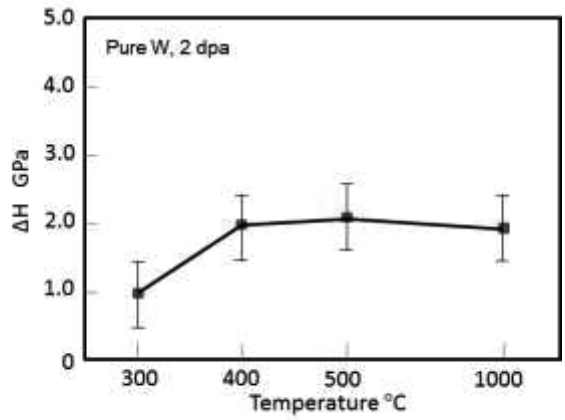

Fig. 2: Irradiation temperature dependence of irradiation hardening in pure-W irradiated to 2 dpa.