Stress Corrosion cracking susceptibility of reduced-activation martensitic steel F82H

Y. Miwa\textsuperscript{a}, S. Jitsukawa\textsuperscript{b} and T. Tsukada\textsuperscript{b}

\textsuperscript{a}Nuclear Energy and Science Directorate, Japan Atomic Energy Agency, 2-4 Shirakata-shirane, Tokai-mura, 319-1195 Ibaraki-ken, Japan

\textsuperscript{b}Japan Atomic Energy Agency, Shirakata-Shirane 2-4, Tokai-mura, Naga-gun, 319-1195 Ibaraki-ken, Japan

miwa.yukio@jaea.go.jp

For fusion power source in near future, supercritical water-cooled type blanket system was planned in Japan Atomic Energy Agency. The blanket system was designed by the present knowledge base and a reasonable extrapolation in material and design technology. Reduced-activation martensitic steel, F82H, is one of the blanket system structural materials. Therefore durability of the F82H for corrosion and stress corrosion cracking (SCC) is one of the concerns for this water-cooling concept of the blanket system. In this paper, SCC susceptibility of F82H was studied after heat treatments simulating post weld heat treatment (PWHT) or neutron-irradiation at 493K to a dose level of 2.2 dpa. In order to evaluate SCC susceptibility of F82H, slow strain rate testing (SSRT) in highpurity, circulating water was conducted at 513-603 K in an autoclave. The strain rate was $1.0 \times 10^{-7} \text{ s}^{-1}$. Concentration of dissolved oxygen and hydrogen of the circulating water was controlled by bubbling with these gases. Specimens were heat treated after normalization at 1313 K for 40 min and water quenching. Some of the specimens were tempered at 873-1073 K for 1 h. Since the temperature control during PWHT in vacuum vessel by remote handling will be difficult, it is expected the tempering temperature will be different at place to place. Some specimens after tempering at 1033 K for 1 h were irradiated at 493 K to 2.2 dpa in Japan Research Reactor No.3 at Japan Atomic Energy Agency. The SSRT results showed the as-normalized specimens failed by IGSCC in oxygenated temperature water at 573 K. SSRT results of specimens with other tempering temperature conditions will be presented at conference. In irradiated specimen, IGSCC did not occur in oxygenated water at 5113-603 K. IGSCC also did not occur in hydrogenated water at 573 K. However TGSSC occurred in the irradiated specimen with a round notch (radius= ~0.2mm) in oxygenated water at 573 K. The TGSSC was observed at compress stress field in the notched specimen. At notch root, flat ductile fracture surface was observed. These fracture modes were not observed in unirradiated, notched specimens. The effects of notch will be discussed in the paper.