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SCIENCE DIRECT
Journal of Nuclear Materials 340 (2005) 219–236
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Mechanism of irradiation assisted stress corrosion crack initiation in thermally sensitized 304 stainless steel

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Received 24 February 2004; accepted 26 November 2004

Abstract

Thermally sensitized 304 stainless steels, irradiated up to 1.2×10^{21} n/cm^2 ($E > 1$ MeV), were slow-strain-rate-tensile tested in 290 °C water containing 0.2 ppm dissolved oxygen (DO), followed by scanning and transmission electron microscopic examinations, to study mechanism of irradiation-assisted-stress-corrosion-crack (IASCC) initiation. Intergranular (IG) cracking behaviors changed at a border fluence (around 1×10^{20} n/cm^2), above which deformation twinning were predominant and deformation localization occurred earlier with increasing fluence. The crack initiation sites tended to link to the deformation bands, indicating that the crack initiation may be brought about by the deformation bands interacted with grain boundaries. Thus the border fluence is equivalent to the IASCC threshold fluence for the sensitized material, although the terminology of IASCC is originally given to the non-sensitized materials without microstructural definition. The IASCC threshold fluence was found to change with irradiation conditions. Changes in IASCC susceptibility and IASCC threshold fluence with fluence and DO were further discussed.

PACS: 61.72.Lk; 61.72.Mm; 61.72.Nn; 61.80.HG; 61.82.Bg; 62.20.Fe; 68.37.Hk; 68.37.Lp

1. Introduction

Austenitic stainless steels (SSs) used for the light water reactor components tend to suffer from degradation in the environment of radiation and high temperature water during long-term service. Two kinds of material degradation phenomena, namely, irradiation assisted stress corrosion cracking (IASCC) and intergranular stress corrosion cracking (IGSCC) for the irradiated, non-sensitized SSs and the non-irradiated, thermally sensitized SSs, respectively in water environment are known, although their behaviors are similar to each other. Their primary cause has been attributed to the grain boundary chromium (Cr) depletion, produced either by welding during fabrication for IGSCC [1], or by radiation induced segregation (RIS) during service for IASCC [2]. The Cr depletion theory however is nowadays uncertain due to the facts that austenitic 304L and/or 316L SSs having no grain boundary Cr depletion are unable to prevent IGSCC in the BWR water environment [3]. Although premature failure by intergranular environmental cracking of materials exposed to ionizing irradiation has originally termed as IASCC [2] and many metallurgical

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0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2004.11.012

factors such as radiation hardening, ductility loss, RIS of alloying elements of Cr and Ni and residual elements at grain boundaries and radiation induced microstructural evolutions appear to be influential, it is essential for mechanistic understanding of IASCC to identify what the mechanism of intergranular (IG) crack initiation, and how metallurgical factors noted above affect IG crack initiation. In this regards, a recent work of Busby et al. [4] on post-irradiation annealing of proton-irradiated austenitic SSs showed that the IG cracking susceptibility in BWR water changed while leaving RIS virtually unchanged, indicating that Cr depletion is not the primary determinator for IASCC. On the contrary Fukuya et al. [5] more recently insisted that grain boundary segregation, probably Cr depletion, was sufficient to cause IASCC in oxygenated water, based on microstructural observation, grain boundary compositional analyses and slow strain rate tensile (SSRT) tests in 561K oxygenated water for solution annealed SUS 304L, 316 and 316LSSs irradiated to 0.8 dpa in a material test reactor. The above two works apparently suggests that the role of Cr depletion on the mechanistic process of IASCC has not decisively been proved.

Presence of neutron fluence threshold for IASCC at about 5×10^{20} n/cm^2, corresponding to about 0.7 dpa, for the solution annealed austenitic SSs in BWR condition has been a consensus in the IASCC study community [6]. This threshold fluence has been determined phenomenologically from post-irradiation laboratory tests that exhibit a characteristic rise in susceptibility, in addition to from field experiences, without considering mechanistic IG crack initiation process of the irradiated austenitic stainless steels. It is unsuccessful to ascribe the presence of the threshold fluence for IASCC to the RIS-induced chromium depletion. It is still uncertain what the IASCC threshold fluence actually implies, and what the mechanistic process of IASCC initiation is. Therefore the objective of this work is to identify mechanistic meanings of the fluence threshold for IASCC by examining tensile properties, IG cracking responses, fractographic and microstructural features as a function of neutron fluence. Attention is focused on mechanical aspects of IG crack initiation in oxygenated water, since it has been proved that IG cracking can occur without involvement of water environment on the surface region of the irradiated, thermally sensitized 304 SS [7,8]. The presence of mechanism mechanically causing IG cracking in inert gas in the irradiated, thermally sensitized 304 SS suggests that even if IG cracking occurs during SSRT tests in water environment, water itself is not absolutely effective for nucleating and initiating IG cracks. Based on mechanistic consideration of IG crack initiation process in inert gas, we hypothesize that high stress and strain concentrations at grain boundaries are more essential than the water environment for IG crack initiation [8].

2. Experimental

The tensile specimens were fabricated from a solution annealed 304 SS tube having an outer diameter of 10 mm and a thickness of 1 mm. The shape and dimension of the specimen are shown in Fig. 1. The gauge section of the tensile specimen is 16 mm long and 3 mm wide, and one side of the specimen is convex and the other concave. The chemical composition of the unirradiated 304 SSs sample is shown in Table 1. The solution annealed 304 SS specimens were further heat-treated in two steps: at 750 °C for 100 min and then at 500 °C for 24 h. These heat treatments produced thermally sensitized microstructures, that is, a large variety of chromium carbide precipitate at grain boundaries in the 304 SSs, resulting in large changes in chemical compositions, especially Cr depletion at the grain boundary. The concentrations of major alloying elements at grain boundaries of the thermally sensitized material, measured by field emission typed transmission electron microscopy, were Cr: 11.0, Ni: 12.0 and Fe: 75.2 wt%.

The thermally sensitized 304 SS tensile specimens were irradiated to neutron fluences ranging from 7.5×10^{19} to 1.2×10^{21} n/cm^2 ($E > 1$ MeV) in test reactors. Table 2 summarizes the irradiation conditions of the specimen. All the specimens were irradiated in the capsules filled with helium gas, except for those of the highest fluence, which were irradiated in an environment of oxygenated pure water. Neutron irradiation in water apparently formed thicker specimen surface oxides films than that in inert gas did. There is however no clear evidence for the irradiation environment of oxygenated water to have affected the occurrences of IG cracking in this work. It has been demonstrated that neutron irradiation to around 10^{21} n/cm^2 causes a reduction in grain boundary chromium concentration by only a few % for the thermally sensitized 304 SS [9].

A SSRT technique was applied for the tensile tests of the unirradiated and irradiated specimens at 290 °C in inert gas and in oxygenated water containing 0.2 ppm dissolved oxygen (DO) at the CIEMAT hot cell facility.

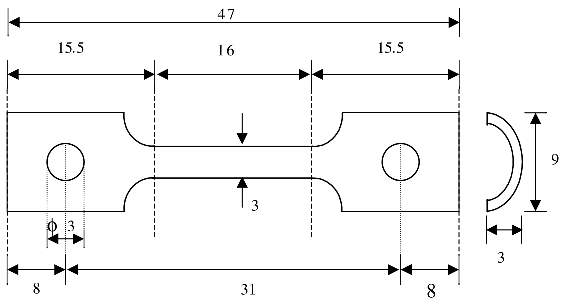

Fig. 1. Shape and dimension of the tensile specimen.