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Corrosion fatigue behavior of Alloy 690 steam generator tube in borated and lithiated high temperature water

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ABSTRACT

Corrosion fatigue behavior of Alloy 690 tube used for actual steam generator (SG) was investigated in <5 ppb (by weight) dissolved oxygen borated and lithiated high temperature water by low cycle fatigue tests with boat-shaped specimens. It was found that the present tube materials had longer fatigue life than round-bar materials in open literature. The surface cracks were tortuous, typical crack branching and linkage were observed. The fracture surfaces were rough with well-defined fatigue striations. Related mechanisms of fatigue crack initiation and propagation of Alloy 690 SG tube in high temperature water are discussed.

1. Introduction

Nickel-based Alloy 690 is an important structural material for steam generators (SGs) in nuclear power plants (NPPs). Though failures of Alloy 690 SG tube have rarely been reported, it may suffer from environmental degradation such as pitting, stress corrosion cracking (SCC) and corrosion fatigue (CF). The flow induced turbulence and thermal loading due to thermal stratification and thermal striping may provoke fatigue damage to the SG tubes. Some failures of SGs have been attributed to CF [1]. Therefore, SG tube CF failures can become a significant damage mechanism impacting SG reliability, especially as the units become older. Many laboratory testing results [2–11] also indicated that the effects of light water reactor (LWR) environments could significantly reduce the fatigue life of structural materials. However, such an environmental degradation effect was not fully addressed in the current ASME code design fatigue curves. Therefore, the Regulatory Guide 1.207 [12] issued by US NRC required a new NPP have to incorporate the environment effects into fatigue analyses. Different proposals [3,4] now exist for incorporating environmental effects, including temperature, dissolved oxygen (DO), strain rate, strain amplitude and sulfur content, into the fatigue design curves for nuclear grade structural materials. The results indicated that LWR environments decreased the fatigue life of nickel-based alloys prominently, although the extent of the environmental effects was considerably less than the austenitic stainless steels. Higuchi et al. and Hong et al. [4,7,9] also reported that the PWR primary water decreased the fatigue life of Alloy 690. However, the fatigue life model for the nickel-based alloy was based on limited fatigue data and there is few published literature for CF behavior of nickel-based alloys in PWR environment, especially for Alloy 690. In addition, most of the fatigue data for Alloy 690 in open literature was obtained by fatigue tests with round-bar specimens, and little work was based on actual SG tube. The results from round-bar material documented in the open literature is not representative of actual behavior of SG tube due to the mutually interactive influences of microstructural features, section thickness and processing variables.

Meanwhile, it is of great significance to understand the mechanism of enhanced fatigue crack initiation and propagation of Alloy 690 in LWR environments. It is generally accepted that the enhanced crack growth rate of nuclear grade structural materials in LWR environments could be attributed to either slip oxidation/dissolution or hydrogen induced cracking [9–11,13–18]. Chopra and Shack [3] suggested that environmentally assisted cracking reduction in fatigue lives of austenitic alloys most likely was not caused by slip oxidation/dissolution but some other process, such as hydrogen-induced cracking. Hong et al. [9] reported the accelerated fatigue crack initiation and growth of Alloy 690 in PWR environment as a combination of metal dissolution, hydrogen produced by corrosion reaction and dynamic strain

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aging. Seifert et al. [5] insisted that the enhanced fatigue crack growth rate was attributed to slip oxidation/dissolution. Therefore, there still exists much controversy regarding to which one is the dominant mechanism in the process of fatigue crack initiation and propagation in LWR environments.

The present paper designed a kind of boat-shaped specimen, which was a part of actual Alloy 690 SG tube, and corresponding fixtures for tension–tension low cycle fatigue (LCF) tests of actual Alloy 690 SG tube in high temperature water. The LCF behavior of two types of as-received Alloy 690 SG tubes was investigated in borated and lithiated high temperature water (325 °C). The present test data is compared to ASME code design fatigue curves and ANL fatigue design model, and the possible mechanisms for fatigue crack initiation and propagation are discussed.

2. Materials and experimental procedure

2.1. Materials

Two types of as-received Alloy 690 SG tubes were used in the present work. One was made in Japan, which was labeled as tube J. The other one was made in China, which was labeled as tube C. Their compositions and mechanical properties are listed in Table 1 and 2. Both tube J and tube C consist of near equiaxed grains and discrete carbides $M_{23}C_6$ with average grain size of about 30 μm and 25 μm, respectively (Fig. 1).

2.2. Specimen and experimental procedure

The diameter and thickness of the as-received SG tubes are 17.4 and 1.01 mm, respectively. Because of its thickness is too small to design standard fatigue specimen according to American Society of Testing and Materials Standard (ASTM) E606-04, the designed fatigue specimen mainly based on principle of keeping its stability during fatigue tests by a series of repeated experiments. A kind of boat-shaped LCF test specimen with 8 mm gauge length, 6.3 mm gauge width and 1.01 mm thickness was designed and machined along the axial direction of the SG tubes (Fig. 2). The specimen kept the original intrados and extrados surfaces of the SG tubes and its lateral surfaces (machined surfaces) were ground successively with silicon carbide paper up to 2000 grit, which made sure its surface roughness (Ra, which is the arithmetical mean deviation of the profile) was better than 0.4 μm. Then, the specimen was cleaned ultrasonically with alcohol, dried and preserved in a desiccator. A matched fixture was designed and employed to grip the boat-shaped fatigue specimen. Repeated fatigue testing results showed the fixture could provide stable holding force, and the fatigue specimen could stay steady during strain-controlled tension–tension fatigue testing. The equipment for LCF tests in high temperature water consisted of an electro-servo hydraulic fatigue testing machine of ±50 kN in dynamic load, an austenitic stainless steel autoclave of 10 L in capacity and a water loop with 15 L per hour in flow rate, and a linear variable differential transformer (LVDT) system was employed to in-situ monitor the strain of the specimens' gauge length in high temperature high pressure water (Fig. 3). Fatigue tests were performed in an axial stroke control mode with triangular waveform. The test conditions and water chemistry are summarized in Table 3. The solution was

Table 1
Composition of the investigated Alloy 690 SG tubes (wt%).

	Ni	Cr	Fe	Mn	C	Al	Ti	Cu	Si
Tube J	59.3	29.89	9.2	0.26	0.018	0.13	0.26	0.029	0.31
Tube C	60.2	29.47	9.8	0.018	0.020	0.11	0.13	0.010	0.04

Table 2
Mechanical property of the investigated Alloy 690 SG tubes.

	$R_{p0.2}$ (MPa)	R_m (MPa)	Elongation (%)
Tube J	321	732	67.8
Tube C	355	779	62.1

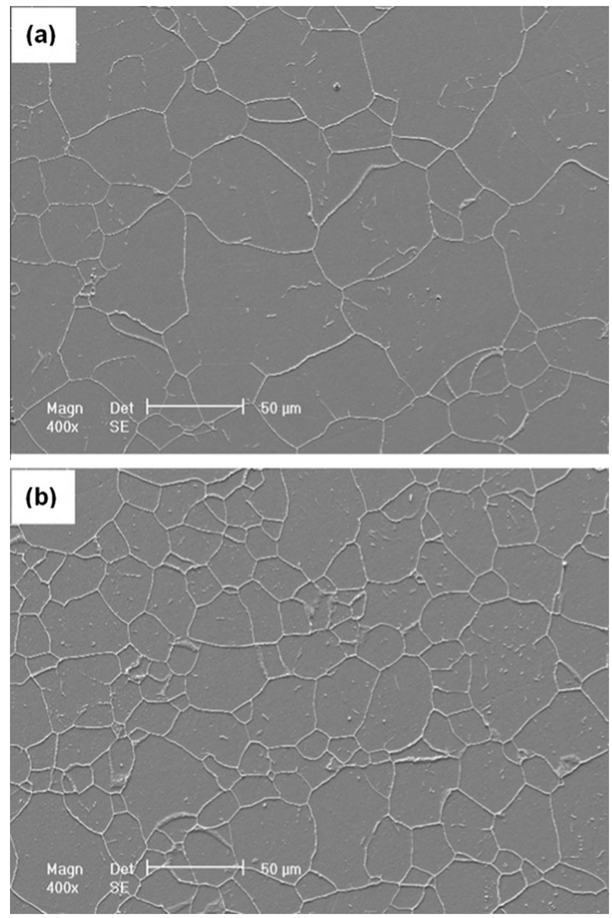

Fig. 1. SEM morphologies of microstructure of tube J (a) and C (b).

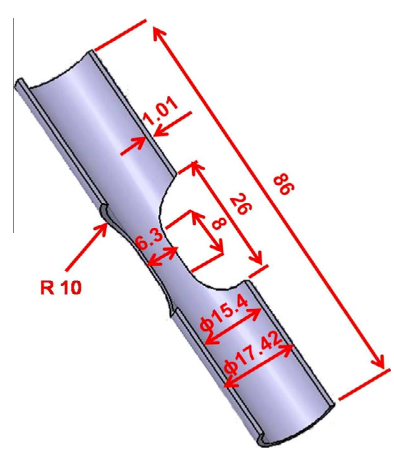

Fig. 2. Illustration of boat-shaped specimen used in the present work.

2.2 ppm (by weight) Li as $LiOH{\cdot}H_2O$ and 1200 ppm B as H_3BO_3 in the present work. DO concentration of the inlet water was monitored with METTLER TOLEDO Thornton Model 3X7-210 DO