Effect of thermal aging on Low Cycle Fatigue of Alloy 690

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Abstract : Thermal aging is one of the concerns for the long-term operation of nuclear power plants. Indeed, components in the primary circuit undergo thermal aging while exposed to the chemically active environment of Pressurized Water Reactors (PWRs) over time. Among the materials used in the reactor components, Alloy 690 can be found in some critical components for nuclear safety. Despite its importance, research on the effect of thermal aging on the microstructural changes and low cycle fatigue (LCF) behavior of Alloy 690 remains limited. This study aims to assess the impact of thermal aging on the fatigue life of Alloy 690. The as-received sample underwent aging at 420°C for 4000 hours, representing the equivalent aging of 60 years in reactor working conditions. First, the characterization of the area and density of intergranular and intragranular precipitates was performed to understand the microstructural changes in the aged specimen. Then, low cycle fatigue tests were conducted on the as received and aged samples at varying strain amplitudes. To investigate the influence of thermal aging on the fatigue behavior of Alloy 690, fracture surfaces were analyzed to estimate fatigue crack growth rates based on striation spacing measurements. Additionally, the axially cut fractured samples have undergone analysis using Electron Backscatter Diffraction (EBSD) to understand the effect of aging on strain localization near the crack path. Results indicate that while the characterization of the area and density of intergranular precipitates in the aged specimen (for 2000 hours, approximately 30 years) showed no significant changes, there was a slight increase in the area and density of intragranular precipitates under the same conditions.

Keywords : alloy 690, thermal aging, low cycle fatigue, precipitates
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