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Investigation of failure behavior of two different types of Zircaloy clad tubes used as nuclear reactor fuel pins

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ABSTRACT

Nuclear reactor fuel pins act as barriers to the release of radioactive fission products to the coolant flowing around these thin-walled tubes and hence they prevent the leakage of radioactivity to the surroundings of reactor core. These tubes are of small thickness in order to have less resistance in the path of heat flow from the fuel to the coolant. Investigation of failure behavior of these fuel clad tubes is of utmost importance to the designers and plant operators in order to ensure the maximum residence time of the fuel bundles inside the reactor core as well as to ensure minimal activity during operation and refueling activities. Various types of zirconium based alloys are used to manufacture these pins. The focus is to obtain better strength, ductility, corrosion resistance, oxidation resistance, and minimal creep including those due to irradiation-assisted damage and deformation processes. Two number of such types of alloys, namely, re-crystallization annealed Zircaloy-2 and stress-relief annealed Zircaloy-4, have been investigated in this work for their fracture behavior. As standard fracture mechanics specimens cannot be machined from these thin-walled tubes, non-standard specimens with axial cracks have been used in this work. Load normalization technique has been used to evaluate crack growth during loading of these specimens. It was observed that the re-crystallization annealed Zircaloy-2 specimens have higher initiation fracture toughness as well as higher resistance to crack growth compared to the other type of specimens. In order to understand the micro-structural aspects of the fracture resistance behavior of these materials, further investigation incorporating optical and transmission electron microscopy have also been carried out. It was concluded that the higher fracture resistance behavior of the re-crystallization annealed Zircaloy-2 specimens can be attributed to the presence of finer grain and sub-grain micro-structure, very low dislocation density and other defects in the material.

1. Introduction

Zirconium alloys are widely used as the fuel cladding tubes and other core internals in light water reactors such as boiling water reactors (BWR), pressurized water reactors (PWR) as well as pressurized heavy water reactors (PHWR). These alloys have favorable properties such as high mechanical strength, good ductility, very low neutron absorption cross-section and good corrosion resistance. A review of the development and application of these alloys is presented in Ref. [1].

The development of zirconium alloys for nuclear reactor applications started with addition of 2.5% Sn to pure zirconium producing Zircaloy-1 which had good corrosion resistance, strength, and formability. Later, addition of Fe and Cr to Zircaloy-1

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