Chapter 9

FE Analysis and Experimental Investigation of Cracked and Un-Cracked Thin-Walled Tubular Components to Evaluate Mechanical and Fracture Properties

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ABSTRACT

For investigation of fracture behavior and assessment of remaining life of critical thin-walled tubular components in industry, the transverse mechanical property and the axial fracture properties are essential. However, evaluation of these properties by machining suitable standard specimens from these components directly and subjecting them to standard tests is usually not feasible. In this chapter, the nonlocal version of the Rousselier’s damage model has been used to predict the fracture resistance behavior of double-edged-notched-tensile specimens made from Zircaloy-4 material. Initially, the micromechanical parameters have been determined from the testing of ring-type specimens. Subsequently, these parameters were used in finite element analysis of the double-edged-notched-tensile specimen in order to predict the crack growth behavior and the crack path under applied displacement-controlled loading conditions. The fracture resistance behavior obtained in terms of J-R curve was also compared with the corresponding J-R curves of axially-cracked pin-loading-tension specimens.

INTRODUCTION

Zirconium alloys are used for manufacturing of various types of components in nuclear reactors, such as fuel-clad tubes, pressure tubes, spacer grids etc., because of the excellent mechanical properties, corrosion resistance behavior and low neutron absorption cross-section of these materials. In order to design for maximum burn-up and higher resident times of fuel-bundles in the nuclear reactor, one needs to know the

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fracture properties along with the mechanical properties (Leclercq et al., 2008; Link et al., 1998; Daum et al., 2002). Unlike mechanical properties, the fracture initiation toughness and the fracture-resistance curves depend upon the geometry of the components (Needleman & Tvergaard, 1984; Kussmaul et al., 1995; Kim et al., 2016). The fracture behavior of a thin-walled tube is significantly different from that of a thick-walled tube and a thick-walled standard fracture mechanics specimen. Experiments are usually conducted on standard fracture mechanics specimens (thickness being of the order of 25 mm) in the laboratory to determine the J-R curves. However, these thick-walled specimens cannot be machined from the thin-walled tubes, such as nuclear reactor fuel-clad tubes, where the thickness is of the order of 0.4 to 0.9 mm.

It may also be noted that these tubes undergo various mechanical working and heat-treatment processes during their manufacturing process and hence, the fracture properties of as-manufactured fuel-clad tubes are important from the point of view of designers and safety analysts. The main priority in the nuclear industry is the safe, reliable, and economic operation of different types of nuclear reactors. Recently, the development of accident-tolerant fuel has become very important after the 2011 Fukushima accident (Kim et al., 2006, 2007, 2008, 2013, 2016; Youinou & Sen, 2014; Zinkle et al., 2014; Ott et al., 2014; Terrani et al., 2014). The emphasis is to increase fuel safety and reliability during normal operations, operational transients, and also accident events. The microcell UO₂ and high-density composite pellet concepts are being considered in order to achieve enhanced retention capabilities of highly radioactive and corrosive fission products within the fuel-pellets (Kim et al., 2016; Youinou & Sen, 2014; Zinkle et al., 2014). High-density pellets are being planned to be used along with new types of cladding, e.g., surface-modified Zr-based alloy (Kim et al., 2016), SiC composite material (Kim et al., 2013; Stone et al., 2015; Stempien et al., 2013), and oxidation-resistance iron-based alloys (Terrani et al., 2013; Pint et al., 2013) are the important concepts which are being pursued by researchers which can effectively suppress hydrogen explosions and the release of radio-nuclides into the environment.

For safety-critical components made up of ductile materials (e.g., Zircaloy, ferritic pressure vessel steel, austenitic stainless steel, aluminum alloys etc.), finite element (FE) approaches which use material damage constitutive models (Needleman & Tvergaard, 1984; Kussmaul et al., 1995; Pitard-Bouet et al., 1999; Rousselier, 1987; Gurson, 1977; Tvergaard & Needleman, 1984; Rudnicki & Rice, 1975) are extremely useful in predicting the failure behavior and the corresponding process with high accuracy. For structural integrity assessment accounting for design-basis accident conditions (such as loss-of-coolant-accident and reactivity-insertion-accidents), the fracture resistance data in terms of J-R curves are also required to be evaluated at different temperatures.

Continuum damage mechanics approaches (Rousselier, 1987; Gurson, 1977) offer an impressive alternative to predict the J-R curves of the components with various postulated cracks and loading conditions. In the last two decades, damage mechanics based models like Rousselier’s model (Rousselier, 1987), Gurson’s model (Gurson, 1977; Tvergaard & Needleman, 1984) have become powerful tools in the safety analysis of nuclear reactor components, e.g., pressure vessels, shell-nozzle junctions, pipings, headers and elbows etc. The above reactor components are usually made up of low-alloy ferritic grade steels and austenitic stainless steels. However, the fuel-clad tubes and pressure tubes are made up of Urumium-based alloys and the study of their fracture behavior in the context of damage mechanics has not received much attention in literature.

These damage models predict the material behavior on the basis of the micromechanical processes (i.e., void nucleation, growth and coalescence) leading to ductile fracture. In ductile materials, voids tend to nucleate in the regions of plastic strain localization in the presence of dominant hydrostatic stresses.