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Characterization of Creep-Fatigue Crack Growth in Alloy 709 and

Prediction of Service Lives in Nuclear Reactor Components

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Program: Nuclear Reactor Technologies, Creep and Creep-Fatigue Crack Growth Mechanisms

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ABSTRACT:

Project Objectives

Austenitic stainless steel Alloy 709 is a fairly recent alloy developed for high temperature structural applications in power plants. It exhibits excellent creep strength, corrosion resistance and it is being considered alongside other steels as a potential material of choice in structural components for Very High Temperature Gas Cooled Reactors, Sodium Cooled Fast Reactors and Fluoride Salt Cooled High Temperature Reactors. It is essential for the nuclear engineers to have access to detailed databases and predictive computational models for all important properties of Alloy 709, including creep-fatigue crack growth behavior, in order to compare its pros and cons versus other materials. This will allow for informed decisions to be made concerning selection of appropriate materials for nuclear reactor applications. The proposed project aims to address the research needs outlined in Department of Energy programmatic documents. Specifically, the project objectives are:

- To measure and report creep-fatigue crack growth rates in Alloy 709 under various creep-fatigue loading in accelerated test conditions and to use this knowledge to predict the extended service life of structural components.
- To measure and report the creep-fatigue crack growth rates in Alloy 709 in as-received and aged conditions, and to identify extrapolation rules between these conditions.
- To develop and provide a flaw evaluation procedure for creep-fatigue service conditions. The flaw evaluation technique will be validated using predictive computational simulations and experimental testing for validation on several typical crack growth specimens.

Project Description

Experimental measurements of creep-fatigue crack growth rates in Alloy 709 will be performed using compact-tension specimens. Material will be tested in the as-received condition and aged at several relevant temperatures and durations. Crack growth rates will be presented as diagrams of crack growth rates per cycle (for fatigue) and per time (for creep) versus typical fracture mechanics parameters as crack-tip driving forces. The creep-fatigue testing will be performed in accelerated test conditions at elevated temperatures of 650 ^{0}C and 700 ^{0}C and high applied stresses, and in typical service conditions at 550 ^{0}C and low stresses. Microstructure analysis of Alloy 709 specimens will be performed using electron back-scattered diffraction and transmission electron microscopy. The size and distribution of second phase particles and precipitates, and their influence on crack growth will be characterized. Larger particles can act as sources of dislocations in strain gradient plasticity mechanism. Smaller particles will interact with dislocations and influence the deformation kinetics.

Microstructure evolution during aging will be simulated using the thermodynamics codes Thermocalc and TC-Prisma. These codes will be employed to calculate concurrent nucleation, growth/dissolution and coarsening of precipitates, temporal evolution of particle size distribution, average particle radius and number density, volume fraction and composition of precipitates, nucleation rate and coarsening rate, Time-Temperature-Precipitation diagrams and estimation of multi-component interfacial energy. A set of phenomenological equations will be derived from the thermodynamics simulations and used for constitutive modeling in the finite element method.

Computational predictions of crack growth rates in Alloy 709 will be performed with the goal of helping to develop a flaw evaluation procedure for the licensing procedures of nuclear reactors. The compact-tension specimens will be modeled in the finite element method. Simulations of creep-fatigue loading in the actual testing conditions and long term service conditions will be performed to compute the size of process (plastic) zone at the crack tip and crack-tip opening displacements. Strip-yield modeling will be used to simulate creep-fatigue crack growth for all the experimental tests performed in the lab. Strip-yield modeling uses analytical formulations for the stress intensity factors and crack opening profiles given different specimen configuration. Crack growth simulations under accelerated and normal service conditions will thus be readily performed.

A flaw evaluation methodology featuring a combination of experimental testing to monitor the flaw size and computational simulations to predict the flaw size evolution between inspection cycles will be proposed. Three specimen types will be considered for testing and simulations, i.e., the compact-tension, the middle-tension and the round bar specimen. Crack evolution and total failure life will be predicted at temperatures and stresses relevant for nuclear reactor operating conditions. The flaw evaluation method will help toward the licensing activities of materials for nuclear reactors.

Benefits and Outcomes

The characterization of long service life for components manufactured out of Alloy 709 is crucial to the adoption of this material in advanced nuclear reactor applications. Several outcomes will result from this project: 1) *A comprehensive testing database* on the creep-fatigue crack growth in Alloy 709 specimens under a wide range of service conditions. 2) *An understanding* of the underlying mechanisms for crack growth through microstructural characterization and thermodynamic modeling. 3) Equations that will allow extrapolation of crack growth rates from accelerated testing to service conditions and extrapolation from as-received to aged material. 4) *Predictive modeling techniques* of crack growth rates and rupture lives using computational methods. 5) *A flaw evaluation technique for creep-fatigue service conditions*. 6) Results will be disseminated in *research papers* in peer-reviewed journals and conference proceedings.

Major Participants

This project will be a collaboration between researchers from the University of Idaho, Purdue University and the Idaho National Laboratory. Experimental testing and microstructural analysis will be performed at the University of Idaho. Finite element modeling and strip-yield modeling will be performed at the University of Idaho and Purdue University. Thermodynamics simulations will be performed at Idaho National Laboratory. Integration of the experimental and modeling and simulation results into an overall strategy for nuclear reactor applications will be performed by Purdue University. Periodic consultations will take place with researchers from Idaho National Laboratory and Oak Ridge National Laboratory in order to produce the most appropriate set of data and the most relevant research approach that will benefit nuclear engineers.