Development and Integration of Light Water Reactor (LWR) Materials Degradation Codes into Grizzly

PI: Digby D. Macdonald- University of California at Berkeley

Collaborators:
Iouri Balachov 4-D Power
Benjamin Spencer Idaho National Laboratory

Program: Nuclear Reactor Technologies: RC-4: Materials Aging and Degradation.

ABSTRACT:

The primary goal of the proposed, 3-year research program is the development of deterministic models for predicting the accumulation of localized corrosion damage (pitting corrosion, stress corrosion cracking and corrosion fatigue) in the primary coolant circuits of the currently operating fleet of Light Water Reactors (LWRs) and to embed the models into the Grizzly code for modeling nuclear power plant component aging, which is based on Idaho National Laboratory's simulation environment (MOOSE). Localized corrosion in LWR (BWR and PWR) primary coolant circuits is primarily an electrochemical phenomenon, the rate of which is determined by certain electrochemical properties, such as the electrochemical corrosion potential (ECP), solution conductivity, temperature, pH, flow rate, and the kinetics of the reduction of a cathodic depolarizer (e.g. O_2) on the surfaces external to the crack, in addition to mechanical loading (stress intensity factor on the crack) and micro-structural/micro-chemical factors (grain size, precipitates, etc). Because the efficient control of environmentally-assisted cracking (EAC) damage requires the accurate control of these parameter, it is necessary to develop codes that can accurately predict their values and the crack growth rate (CGR) at any point in the primary coolant circuit (PCC) over wide ranges of temperature (25 ^{o}C to 320 ^{o}C), pH (6 – 8), ECP (-0.9 V_{she} to 0.2 V_{she}), solution conductivity, flow rate (1 – 6 m/s), and stress intensity (5 MPa.m$^{1/2}$ – 50 MPa.m$^{1/2}$), reflecting the properties in reactor PCCs. Knowledge of these parameters, along with suitable codes, would allow one to predict the accumulated damage in PCC as a function of the operating history of the reactor. If this can be done successfully, tremendous economic and safety benefits would accrue to the operators, because it would allow the reactors to be operated in a manner, so as to reduce both the risk and cost of damage and to enhance safety. Of particular importance is that the codes would allow operators to explore "what if" options in plant operation, so as to define the resulting cost-benefit and safety-risk issues. We have previously developed codes of this type, DAMAGE-PREDICTOR, ALERT, REMAIN, and FOCUS, for predicting EAC damage in the primary coolant circuits in BWRs. Generally, these codes have demonstrated a remarkable ability to predict EAC damage in operating BWRs, even though they lacked a viable model for crack initiation. A much less capable PWR code (PWR_ECP) was also developed, but it predicted only the ECP, because of a viable coupled environment fracture model (CEFM) for estimating crack growth rate in non-stainless steels, such as Ally 600, was not then available. The proposed work will develop advanced damage simulation codes for predicting EAC damage in the primary coolant circuits of currently operating (Generation II) LWRs; BWRs and PWRs of the generic General Electric and Westinghouse designs, respectively.