

## Assessment of Corrosion Resistance of Candidate Alloys for Accident Tolerant Fuel Cladding under Reactor Conditions

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The environment of a light water reactor (LWR) core is a combination of high temperatures, high pressures, high neutron and gamma fluxes, mechanical stresses and chemical aggressive coolants. All of these factors combined can induce changes in the microstructure of the fuel and cladding that are very difficult to predict in a systematic fashion [1] and are major challenge for the safety and life extension of current reactors. Throughout reactor transients and accidents, the cladding may experience deterioration caused by a temperature increase, oxidation embrittlement [1], [2], or mechanical interaction with the fuel caused by stress [3]. These events may lead to cracking or rupture of the cladding, causing the release of fission products into the coolant. Such events have been observed at the Three Mile Island and Fukushima accidents.

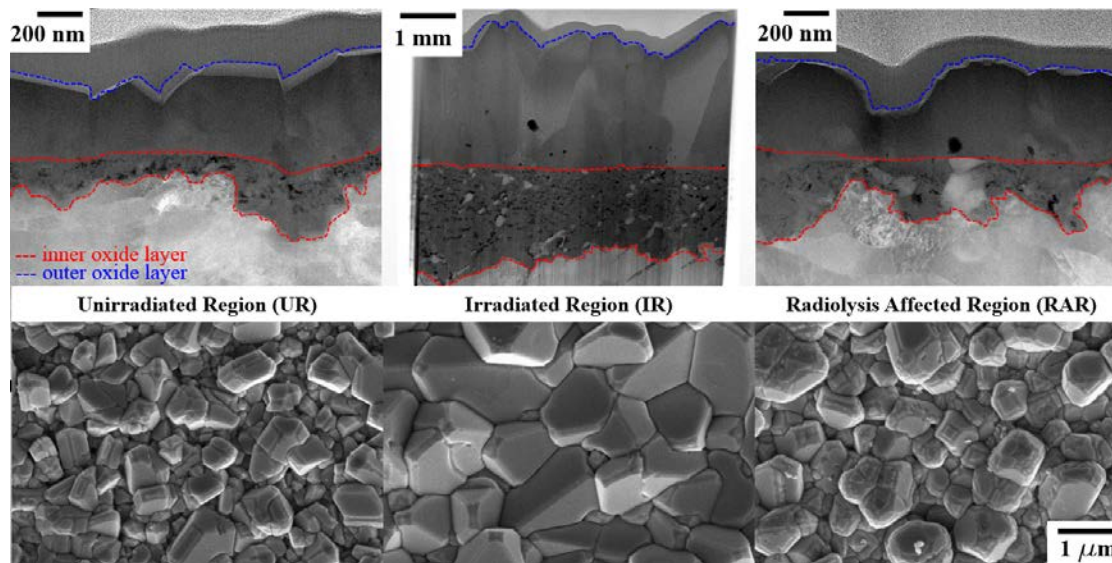
To avoid such occurrences in LWRs, the accident tolerant fuels (ATF) program was initiated to focus on the replacement of zirconium-based alloys with materials that exhibit slower steam oxidation kinetics. This project focuses on several iron-based alloys such as T91, APMT, MA956, experimental Fe-Cr alloys and one experimental nanofeatured alloy (NFA). Experiments have been conducted in both PWR primary water at 320°C and BWR normal water chemistry at 288°C, spanning a large range in electrochemical corrosion potential (ECP). Samples were exposed to either proton irradiation (University of Michigan) or electron irradiation (Notre Dame Radiation Laboratory) to independently assess the roles of displacement damage or radiolysis on the corrosion rate, oxide thickness, morphology, structure and resistivity. Post-irradiation characterization of various regions of the electron-irradiated samples was completed by means of microscopy (Notre Dame Integrated Imaging Facility) and spectroscopy (Notre Dame Materials Characterization Facility) techniques to provide high-resolution information regarding the oxide layer present on the surface of the material.

Under optical microscopy, three distinctive flow regions were identified on the sample surface, Irradiated Region (IR), Radiolysis Affected Region (RAR) and Unirradiated Region (UR). Surface oxide morphology was analyzed via high definition scanning electron microscope (SEM) images. Focused ion beam (FIB) lift-outs from the various regions were prepared for transmission electron microscopy (TEM) analysis of the oxide layer thickness as well as energy-dispersive X-ray (EDX) line profiles were collected to reveal elemental distribution information about the oxides. Post-irradiation characterization images are shown in Figure 1 and Figure 2.

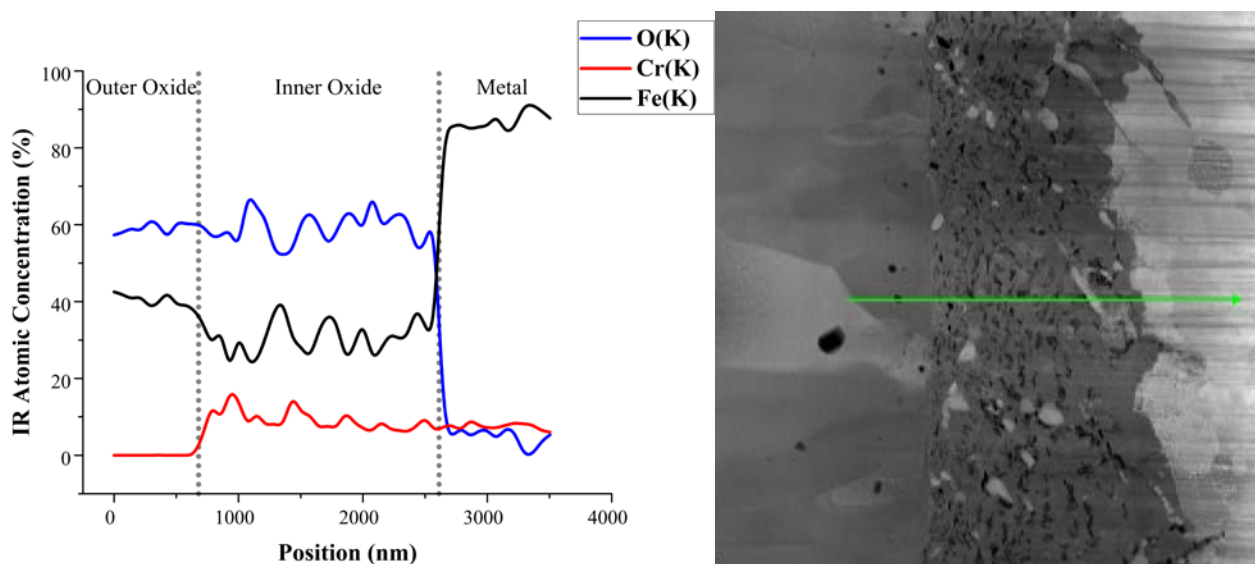
The accelerated corrosion experiments will allow us to determine the selected material's performance under the conditions of extreme radiation in either hydrogenated or oxygenated water at both high temperature and high pressure.

## References:

- [1] M. Pukari and J. Wallenius, "Cladding materials and radiation damage," Dep. React. Phys. K. Tek. Hogskolan, 2010.
- [2] M. Billone, Y. Yan, T. Burtseva, R. Daum, and others, "Cladding embrittlement during postulated loss-of-coolant accidents," Argonne National Laboratory (ANL), 2008.
- [3] NEA, "Pellet-clad Interaction in Water Reactor Fuels," presented at the CEA Cadarache/DEN/DEC, 2005.



**Figure 1.** Post-Irradiation TEM (top) and SEM images (bottom) of 24-hour electron-beam T91 steel sample in Hydrogen-saturated water. Representation of the three different regions, UR, IR and RAR.



**Figure 2.** Left: STEM EDX line scan results of the 24-hour electron-beam irradiated T91 steel sample in Hydrogen-saturated water, atomic concentration vs. position, representation of outer oxide, inner oxide and metal region of scan. Right: STEM bright field image of irradiated region (the thin arrow shows the EDX line scan direction).