Irradiation-assisted Stress Corrosion Cracking of PWR-irradiated Type 347 Stainless Steel

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ABSTRACT:

This project is intended to support continued operation of light water reactors by furthering understanding of irradiation-assisted stress corrosion cracking (IASCC). In 2016, substantial degradation of Type 347 baffle-former bolts was observed in pressurized water reactors (PWRs) due to IASCC[1]. This unexpected level of degradation resulted in greatly added expense for the affected PWRs due to the need for extended maintenance outages. Such outages are quoted to cost over one million dollars per day and were extended for multiple weeks in order to conduct replacement campaigns of these bolts which are located a fraction of an inch from active fuel assemblies in the reactor core. Such reactive maintenance can be avoided by better predictive capabilities for such degradation, which is one objective of this project. A second objective of this project is to investigate IASCC in a potential alternative PWR coolant additive, KOH. By switching to KOH from the currently used LiOH addition, cost savings as well as a more secure supply chain can be realized. Therefore the objectives of this project are well aligned with the DOE's NE mission, which includes efforts to develop technologies and other solutions that can improve the reliability, sustain the safety, and extend the life of current reactors.

These project objectives will be accomplished using sub-sized 4-point bending samples machined from baffle-former bolts extracted from the same PWRs which experienced the unexpected degradation. The baffle-former bolts to be used in this project are owned by Westinghouse, and currently stored at the Westinghouse Churchill Site hot cell facility where the 4-point bend specimens and transmission electron microscopy (TEM) specimens will be fabricated. The small specimen size will allow multiple nominally identical specimens to be cut from locations in the bolt with equivalent service conditions. As the radiation damage varies predictably between different bolts, and along the length of each bolt, a range of radiation damage levels can be investigated. Only by virtue of this unique and recently tested sample and test design has it been able to obtain multiple data points on initiation as a function of position along the bolt. While this material has been otherwise well characterized in previous failure analysis activities, very limited data is available on the IASCC behavior of such Type 347 steel in controlled conditions.

Upon completion of specimen fabrication the completed samples will be shipped to the University of Michigan Irradiated Materials Testing Laboratory. Here the 4-point bend samples will undergo IASCC testing. Pairs of equivalent specimens will be tested in simulated PWR conditions with either KOH or LiOH water chemistry. Post-test scanning electron microscopy (SEM) examination of the bend specimens, and TEM examination of material with equivalent radiation damage, will allow IASCC behavior as a function of radiation damage and water chemistry to be determined, and also allow crack initiation sites to be correlated with microstructural features of the steel. This will provide engineering benefits to the industry of data to improve predictive models of baffle-former bolt failure and identification of the less aggressive water chemistry, while also providing scientific insight into the IASCC mechanism.

[1] Report from Office of Nuclear Reactor Regulation (NRR) Office, "Degradation of Baffle-former Bolts in Pressurized-Water Reactors," NRC ADAMS Accession Number ML16225A341, October 20, 2016. https://www.nrc.gov/docs/ML1622/ML16225A341.pdf