



**U.S. Department of Energy**

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## **Mitigating IASCC of Reactor Core Internals by Post-Irradiation Annealing**

**PI:** Gary S. Was – University of Michigan

**Co-PIs:** Zhijie Jiao, Yugo Ashida - UM

**Collaborators:** Jeremy Busby, Maxim Gussev – Oak

Ridge National Laboratory

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### **ABSTRACT**

Irradiation-assisted stress corrosion cracking (IASCC) of reactor core internals is a potential lifetime-limiting degradation mechanism for LWRs. It has taken on new urgency with the growing interest in extending operating licenses for the current generation of plants to 60 years or beyond in the U.S. What makes IASCC unique is that it is largely controlled by the persistent damage induced by irradiation. That is, while radiation affects the environment through radiolysis, the onset of cracking in LWR environments is controlled by the radiation-induced persistent defects and damage in the alloy [1]. Post-irradiation annealing (PIA) can partially recover radiation damage, thus potentially mitigating IASCC. Given that code qualification of new alloys could take one to two decades, and that an IASCC-resistant alloy has yet to be identified, PIA is the only viable, near-term hope for ameliorating IASCC in today's LWR fleet.

We will determine the optimum PIA-driven mitigation strategy using well-characterized 304LSS taken from a control rod blade of the Barsebak boiling water reactor in Sweden. The alloy was irradiated to a dose of 5.9 dpa at 288°C and characterized for the irradiated microstructure, RIS, and crack growth behavior [2]. The alloy shows a significant susceptibility to IASCC in BWR normal water chemistry (NWC), and as such, is ideal candidate material for the study of PIA. Our plan is to conduct PIA on ten tensile bars, five round compact tension (RCT) samples, and several blanks that will be machined from irradiated stock by Studsvik, Sweden. Using the literature data [3] as a guide, PIA treatments will be conducted in the temperature range 500-600°C. These treatments are designed to explore the regime where SCC susceptibility is changing. For each annealing condition and the as-irradiated state, the dislocation loop microstructure will be examined using TEM rel-rod dark field technique and RIS at grain boundaries will be characterized using STEM/EDS. Radiation-induced precipitates will be characterized using APT. We will conduct a constant extension rate tensile (CERT) test on one tensile sample in each PIA condition in pure argon at 288°C to preserve the pristine surface for localized deformation characterization, and another in simulated BWR water at 288°C to assess crack initiation susceptibility. Results will allow us to determine: the microstructure features that control IASCC initiation and propagation, the annealing kinetics for each microstructure feature, the roles of the various microstructure features in irradiation hardening, the influence of the various microstructure features on localized deformation, the role of localized deformation on IASCC initiation and propagation, and the optimum mitigation strategy based on PIA.

1. G.S. Was and P. L. Andresen, *Corrosion*, 63 #1 (2007) 19-45.
2. *Crack Growth Rates of Irradiated Commercial Stainless Steels in BWR and PWR Environments*. EPRI, Palo Alto, CA: 2009, report# 1019028.
3. J.T. Busby, G.S. Was and E.A. Kenik, *J. Nucl. Mater.* 302 (2002) 20.