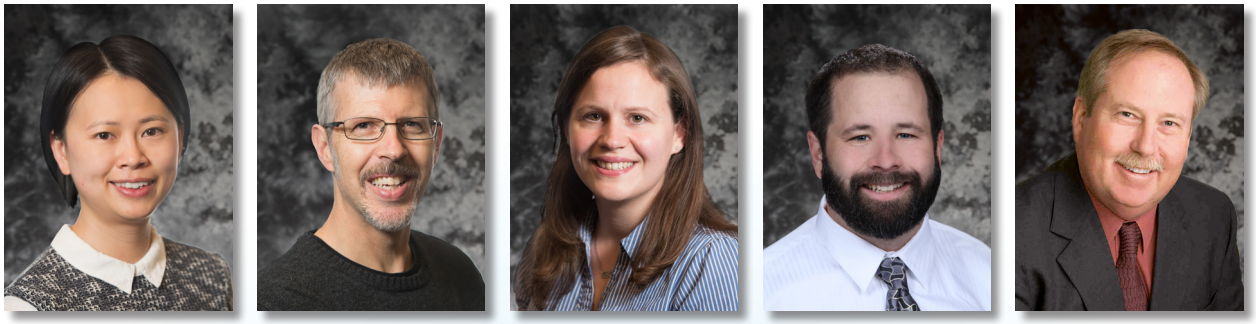
Evaluation of Stress Corrosion Crack Initiation in Nickel-Base Alloys and Implications for PWR Components



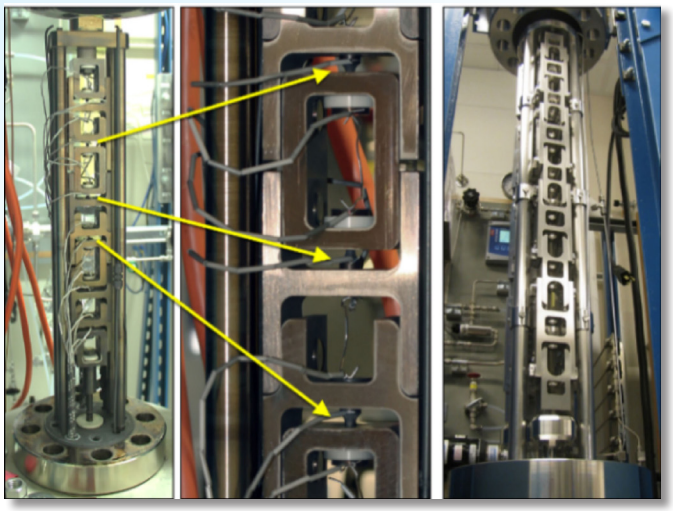
Ziqing Zhai, Mychailo B. Toloczko, Karen Kruska, Matthew J. Olszta, and Stephen M. Bruemmer
Materials Research Pathway

Stress corrosion cracking (SCC) of Ni-base alloys used in light water reactor (LWR) pressure boundary components is a critical issue to the long-term viability of the nation's nuclear fleet. SCC of the originally selected low-Cr Ni-base Alloy 600 and its weld metals used in steam generators and to join piping and instrumentation nozzles to the reactor vessel began to significantly impact pressurized water reactor (PWR) performance in the 1980s and 1990s, which led to their progressive replacement in these components [1]. Although the service performance has been restored through either application of mitigation techniques or replacement with high-Cr Ni-base Alloy 690 and its weld metals, Alloy 600 and its weld metals remain in use in certain regions of the reactor where viable mitigation techniques are still being developed. Meanwhile, SCC susceptibility has been identified in the laboratory for Alloy 690 [2], prompting a need for further assessment of SCC susceptibility for both materials.

This Materials Research Pathway project addresses one of the least understood aspects of SCC for LWR pressure boundary components—crack initiation. Our focus is to investigate important material (e.g., composition, processing, microstructure, strength) and environmental (e.g., temperature, water chemistry, stress) effects on SCC initiation susceptibility of Alloys 600 and 690. The primary objectives are to identify mechanisms controlling crack nucleation, investigate the transition from short to long crack growth under realistic LWR conditions, and establish the framework to effectively model and mitigate SCC initiation.

Three state-of-the-art multi-specimen SCC initiation testing systems were designed and built at Pacific Northwest National Laboratory (PNNL), as shown in Figure 6. The successful implementation of these advanced test systems and methods under the Materials Research Pathway has provided a foundation for SCC initiation studies at the U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) [3], as well as helping to establish the standard for LWR SCC initiation testing around the world [4]. To date, SCC initiation tests have been performed in simulated PWR environments on both Alloy 600 and 690 to evaluate the effects of key material and environmental factors on crack precursor development. For cold-worked (CW) Alloy 600, SCC initiated at the specimen surface following intergranular (IG) attack and grew into the bulk material. In contrast, CW Alloy 690 exhibited internal IG damage in the form of grain boundary cavities, which eventually led to cracks

Figure 6. Small SCC initiation test system with instrumented specimens (left side) and the large SCC initiation test system (right side) located at PNNL.



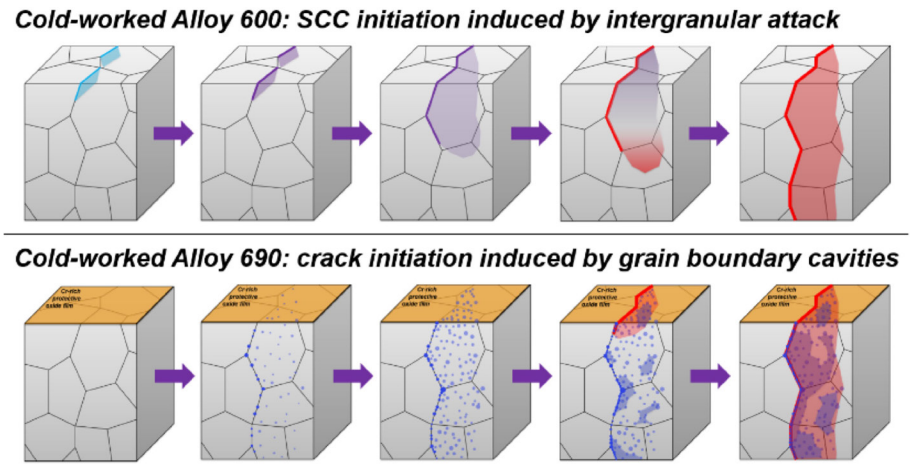
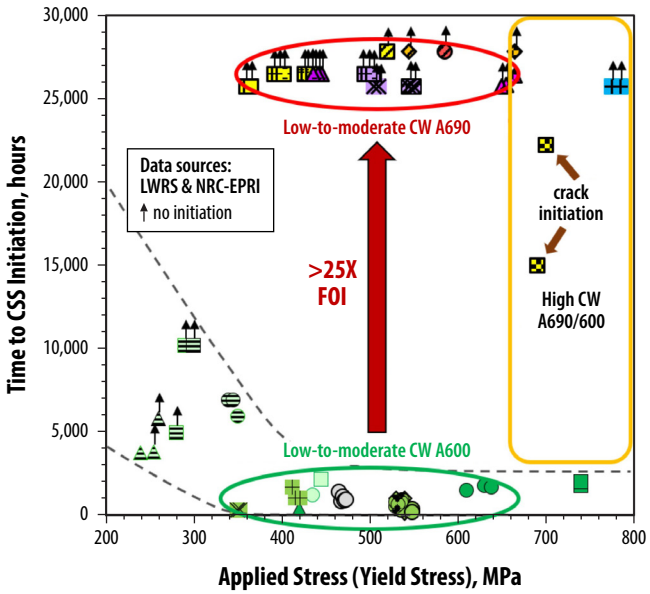


Figure 7. Proposed crack initiation and growth mechanism for CW Alloy 600 (top) and Alloy 690 (bottom) based on experimental observations.

connected to the specimen surface. The different crack initiation and growth mechanisms for Alloys 600 and 690 are illustrated in Figure 7.

SCC initiation data generated for this project and another at PNNL [3] have enabled an estimation of the factor of improvement (FOI) for Alloy 690 versus Alloy 600 in 360°C PWR primary water, as shown in Figure 8. SCC initiation time of less than 1,000 hours was frequently detected in all CW Alloy 600 materials, most of which are in the 15% CW condition. In comparison, SCC initiation has not been detected in any of the low-to-moderate CW Alloy 690 materials surpassing 27,000 hours of exposure at constant load. This suggests the Alloy 690 initiation FOI is greater than 25 times and is still increasing with continued testing. However, crack initiation has been detected in a highly CW Alloy 690 heat after approximately 15,400 and 22,240 hours. All of this information is of critical importance for material degradation prediction and plant life management for existing PWR systems.

Figure 8. Measured SCC initiation time as a function of applied stress for CW Alloy 600 and Alloy 690.



In summary, the ongoing SCC initiation research combining advanced testing and characterization techniques is providing unique insights into the mechanisms and precursor states for SCC initiation in Ni base alloys. This knowledge is enabling the FOI assessment for replacement Alloy 690 and the development of quantitative models to assess the performance of existing Alloy 600 and 690 components. In addition, the basis for improved SCC-resistant alloys and mitigation strategies are being evaluated, all of which are of high interest to the nuclear industry.

References

1. P. M. Scott, P. Combrade, Journal of Nuclear Materials 524, 2019, p. 340-375.
2. S. M. Bruemmer, M. J. Olszta, et al., Corrosion, 69 (10), 2012, p. 953-963.
3. M. B. Toloczko and Z. Zhai, SCC Initiation Testing 2017 Report, PNNL-27011, October 2017.
4. Z. Zhai, M. B. Toloczko, and S. M. Bruemmer, Stress corrosion crack initiation behavior of Alloy 600 and Alloy 690 in PWR primary water, September 2018.