
Investigation of Degradation Mechanisms of Cr-coated Zirconium Alloy Cladding in Reactivity Initiated Accidents (RIA)

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Program: FC-2.5 (NSUF Separate Effects Testing in TREAT using Standard Test Capsule)

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ABSTRACT:

Coating technologies for zirconium-alloy (Zr-alloy) LWR fuel cladding are being actively developed for use as near-term accident tolerant fuel (ATF) cladding. The designs being considered for implementation by Westinghouse Electric Company (WEC) and Framatome include a thin chromium coating on the conventional Zr-alloy cladding. The Cr-coated cladding is expected to provide additional safety margin by virtue of its excellent corrosion/oxidation resistance at accident conditions. Recently, the Pacific Northwest National Laboratory (PNNL) published a comprehensive report on the current status and required safety analysis of the Cr-coated Zr cladding design. The report identifies data gaps for licensing and several performance criteria including performance above Cr-Zr eutectic temperature (1332 °C), post-quench ductility, and coating integrity during swelling/rupture. The recommended qualification data are highly relevant to the transient behavior of the fuel system (e.g., Reactivity Initiated Accidents, RIAs). For accelerated deployment of the Cr-coated cladding in the current U.S. LWR fleet, it is necessary to demonstrate the response of Cr-coated cladding under power transient conditions.

The proposed research aims at investigating of thermal, mechanical, and irradiation response of Cr-coated zirconium-alloy cladding tubes under prototypical Reactivity Initiated Accident (RIA) conditions, in comparison to uncoated Zr-alloy cladding. The objective will be achieved by a pulse-type nuclear heat deposition on the Cr-coated cladding/ UO_2 fuel system followed by comprehensive post-irradiation examination (PIE). Specifically, Cr coatings deposited by two distinctly different methods will be investigated, namely the cold spray (CS) process and the Physical Vapor Deposition (PVD) process. The RIA tests will be performed at the Transient Reactor Test Facility (TREAT) at Idaho National Laboratory, to impose power excursions on the Cr-coated cladding/ UO_2 fuel system in MARCH-SERTTA static water capsule. The target peak cladding temperature (PCT), total energy deposited, and cladding internal pressure will be judiciously selected to demonstrate potential late phase high temperature RIA failures such as cladding ballooning and burst, oxidation embrittlement, and cladding melting. Five primary task areas have been identified in the proposed research work-scope: (1) Sample Fabrication for TREAT Testing, (2) irradiation capsule design and fabrication, (3) TREAT testing for Cr-coated cladding tubes, (4) material characterization and mechanical test after TREAT testing, and (5) analysis of data and modeling of failure mechanism of Cr-coated cladding tubes. The project will be performed collaboratively by the three partnering institutions: University of Wisconsin Madison (UW), University of Illinois Urbana-Champaign (UIUC), and U.S. Nuclear Regulatory Commission (NRC).