Progress in Evaluation of Cr-coated Zr-alloy ATF Cladding under Reactivity-Initiated Accident (RIA) Conditions

Hwasung Yeom [a,b*], Kumar Sridharan [b], Woo Hyun Jung [b], David Kamerman [c]
[a] *Division of Advanced Nuclear Engineering, Pohang University of Science and Technology, Pohang, 37673*
[b] *Department of Engineering Physics, University of Wisconsin, Madison, Madison, WI, 53706*
[c] *Department of Nuclear Fuel Design and Development, Idaho National Laboratory, Idaho Falls, ID USA 83415*
[*] *Corresponding author: hyeom@postech.ac.kr*

***Keywords :** Accident Tolerant Fuel, Reactivity-Initiated Accident (RIA), Cr-coated Cladding*

1. Introduction

Thin chromium (Cr) coated Zr-alloy cladding concepts are being considered for future light water reactors (LWRs) to enhance reactor safety during accidents as well as to achieve extended fuel cycle. However, there are knowledge gaps for robust data sets to support licensing and qualification of these cladding designs under design basis accidents (DBAs) such as reactivity initiated accidents (RIAs) and loss-of-coolant accident (LOCAs).

RIAs are power transients resulting from sudden increase in positive reactivity due to malfunction of control rod mechanism (e.g., ejection or drop in LWRs [1]). As shown in Fig. 1, the uncontrolled widthrawal of a control rod out of the core results in a positive reactivity insertion, allowing rapid increasing local core power. Therefore, the fuel temperature sharply increases, leading to fuel pellet thermal expansion. The reactivity excursion is intially reduced by Dopper feedback and delayed neutron effects followed by reactor trip [2].

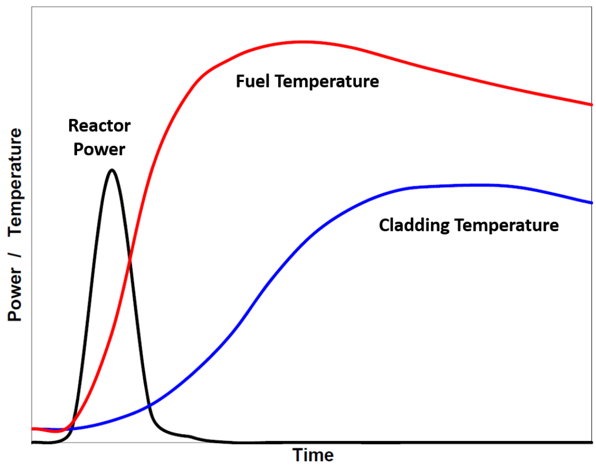

Fig. 1. Illustration of fuel response to a rapid reactivity insertion [2].

The fuel rods are vulnerable to environmental degradation due to high temperature steam oxidation if the fuel cladding is exposed to post-DNB regime during a RIA event. Swelling and rupture failures in the cladding are available due to a combination of high temperature plastic deformation and high fuel rod internal pressure. Rewetting of the damaged cladding would impose significant thermal stress, resulting in potential loss in coolable geometry.

The on-going research project aims at investigating the thermal, mechanical, and irradiation response of Cr-coated Zr-alloy cladding under prototypical reactivity-initiated accidents (RIAs), in comparison to that of uncoated Zr-alloy cladding. This summary provides an overview of the research program funded by U.S. Department of Energy, DOE, including experimental RIA test plan for Cr-coated Zr-alloy cladding tubes using the Transient Reactor Test Facility (TREAT) and the post-irradiation examination (PIE) facility. The experimental data would be a basis for licensing activities of the Accident Tolerant Fuel (ATF) fuel designs.

2. Methods and Results

2.1 TREAT Facility and Test Capsule

The RIA tests have been performed at the TREAT facility at Idaho National Laboratory (INL), imposing a power transient for the Cr-coated cladding/UO_2 fuel system in a HERA static water capsule. TREAT is an air-cooled reactor and the reactor core consists of graphite blocks with a dispersed uranium oxide particle. The graphite core blocks are encapsulated in Zr-alloy sheet metal canisters. The hydraulically actuated transient control rods manipulate various transient power shapes from tens of milliseconds to a few minutes [3]. The TREAT facility is capable of creating truly prototypical pulse widths (30 – 90 ms) for RIAs in PWRs at zero power conditions [3].

The test capsule for the RIA tests was designed and manufactured at INL [4]. The sample tube in the capsule consists of a stack of ten pellets with eight UO_2 pellets and a zirconia insulator pellet on the top and bottom of the stack. During the transient experiment, various types of data are collected such as fuel centerline temperature, fuel cladding temperature, water temperature, boiling detection inside the capsule, and reactor power.

2.2 RIA Test Plan

The RIA experiments were designed to investigate transient fuel behavior in terms of cladding ballooning and rupture as well as the cladding-coolant interaction. The Gaussian pulse energy deposition is in the range of 1000 J/g and 1200 J/g to achieve target peak cladding temperature (PCT). Two PCTs were selected for RIA tests: (i) lower than 1200 °C, the allowable maximum peak cladding temperature of Zr-alloy cladding in LWRs and (ii) above 1200 °C. The PCT can be controlled by the pulse deposition energy in the fuel and the level of overpressure in the HERA capsule. Inert gas is back-filled in the cladding tube (~ 2 MPa) to simulate the accumulation of fission gas inside the cladding. Varying rodlet internal pressure relative to capsule pressure can affect the extent of swelling and rupture of the cladding. Total six RIAs tests were planned and two tests were completed.

2.3 Fabrication of Coated-cladding Samples

Framatome provided nuclear-grade Zircaloy-4 cladding tubes for the Cr coating developments. It is noted that this cladding material has been used extensively for transient tests of LWR fuels in TREAT facility at INL, and as such there is an ample database relevant to this project.

A commercial cold spray deposition system at the University of Wisconsin, Madison was used to deposit feedstock Cr powder on short sections of the Zircaloy-4 tubes. Cold spray deposition parameters were determined to produced ~ 50 μm thick coatings on the tube samples. The coated samples were then polished to decrease the coating thickness down to ~ 25 μm. Three coated samples will be subjected to RIA experiments and some extra samples will be used for room temperature ring compression tests and basic metallurgical characterization. PVD Cr coating on the tube sections have been developed using a magnetron sputtering facility at the University of Illinois Urbana-Champaign (UIUC). Multiple PVD Cr-coated samples will be shipped to INL-TREAT for testing.

2.4 Initial Data of RIA Experiments

Two RIA tests for uncoated, reference Zircaloy-4 samples were conducted (as of August 2023). Fig. 2 shows an as-run reactor power transient shape and measured centerline temperature of a fuel rod sample. The estimated energy deposition was ~ 1000J/gUO_2 and the FWHM pulse was ~ 90 ms. The sample centerline temperature was cut-off at ~2300 ˚C due to the operating temperature limit of the C-type thermocouple. Four thermocouples were installed on the cladding surfaces to measure the surface temperature evolution while there was a large amount of noise due to the TC degradation with the transient. The cladding surface temperature data are still being analyzed.

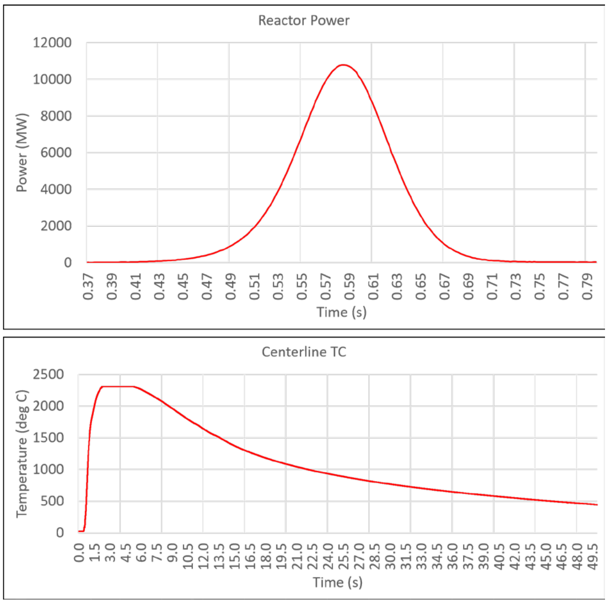

Fig. 2. Measured reactor power pulse and the centerline temperature of uncoated Zircaloy-4 fuel rod with UO_2 during the transient imposed by the TREAT facility.

3. Conclusions

The objective of the project is to perform prototypical RIA experiment for the Cr-coated cladding/UO_2 fuel system followed by comprehensive post-irradiation examination (PIE). The RIA tests has been performed at the Transient Reactor Test Facility at Idaho National Laboratory (INL) to provide power excursions on two types of Cr-coated claddings (cold spray Cr coating, PVD Cr coating). The target peak cladding temperature (PCT), total energy deposition, and cladding internal pressure were chosen to demonstrate potential late phase high temperature RIA phenomena such as cladding ballooning and burst, oxidation embrittlement, and cladding melting. All required samples were already fabricated. Reference tests with uncoated Zr-alloy cladding were completed at TREAT facility and tests for the Cr-coated samples are planned in early 2024. The project has been performed collaboratively by University of Wisconsin, Madison, University of Illinois Urbana-Champaign, U.S. Nuclear Regulatory Commission, and Pohang University of Science and Technology in South Korea.

REFERENCES

[1] D. Kamerman et al., Fuel safety research plan for accident tolerant fuels, *INL/EXT-19-53343*, 2019.

[2] P. Clifford et al., Technical and Regulartory Basis for the Reactivity-initiated Accident Acceptance Criteria and Guidance, *U.S. NRC*, 2015.

[3] N. Woolstenhulme et al., Developing of Irradiation Test Devices for Transient Testing, *Nuclear Technology*, 2019.

[4] H. Yeom at al., Experimental Investigation of Degradation Mechanisms in ATF coated Cladding under Transient Conditions, *ANS Winter Meeting 2022*, Nov. 13-17th, 2022, Phoenix, AZ.