Report Summarizing Boiling Transition (Dryout) Testing of FeCrAl Cladding

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SUMMARY

Nuclear fuel and cladding in a light water reactor must maintain integrity under all normal operational and anticipated operational occurrences. In a Boiling Water Reactor (BWR) some Anticipated Operational Occurrences (AOOs) and Anticipated Transients Without SCRAM (ATWS) events have been known to result in cladding damage due to dryout with conventional Zircaloy-2 claddings. This work presents an assessment of commercially-fabricated, accident tolerant iron-chromium-aluminum (FeCrAl) alloy C26M cladding in such events with regard to the ballooning failure mode for unirradiated cladding

In the severe accident test station (SATS), an experiment has been developed to gauge accident tolerance of fuel cladding materials under simulated cyclic dryout conditions. Thermal cycling from 300°C to 650°C was implemented on C26M and Zircaloy-2 tubes. The C26M tube did not fail after 54 cycles. The Zircaloy-2 tube run under the same parameters as the C26M tube failed during the fourth cycle, consistent with expectations from the literature. Due to thermal gradients present in the furnace, both tubes experienced a maximum temperature at ~780°C. Due to differences in tube geometry, the hoop stress of the C26M tube was significantly higher than that of the Zircaloy-2 tube, 76 MPa at the maximum peak temperature compared to 55 MPa at burst, respectively. The cyclic dryout experiments show an increased performance of C26M over Zircaloy-2 under simulated conditions, an indication of C26M possessing superior high temperature oxidation resistance, mechanical properties and potentially resistance to fatigue.

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FIGURES

ACRONYMS

AOO	Anticipated Operational Occurrence
ATF	Accident Tolerant Fuel
ATWS	Anticipated Transient Without SCRAM
BWR	Boiling Water Reactor
ORNL	Oak Ridge National Laboratory
SATS	Severe Accident Test Station

REPORT SUMMARIZING BOILING TRANSITION (DRYOUT) TESTING OF FECRAL CLADDING

1. INTRODUCTION

Boiling water reactor (BWR) fuel cladding failure can occur due to dryout during anticipated operational occurrences (AOOs) or Anticipated Transients Without SCRAM (ATWS) events [1-3]. Boiling transition is an unstable thermal-hydraulic phenomenon occurring on the heated surface at a heat flux between the maximum attainable in nucleate boiling and the minimum attainable in film boiling. Boiling transitions on the fuel cladding surface above a limiting temperature in accident scenarios are likely to introduce dryout, or cyclic dryout and rewetting. One of the fundamental goals of BWR operation is to avoid fuel system failure due to AOOs or ATWS events.

The limits for cladding failure due to sustained dryout or cyclic dryout and re-wetting are relatively well-understood for zirconium-based cladding (e.g. Zircaloy 2). However, Accident Tolerant Fuel (ATF) cladding materials, such as iron-chromium-aluminum (FeCrAl) alloys, have the potential to withstand much longer dryout durations due to their improved oxidation resistance and enhanced mechanical properties relative to Zr-based claddings [4-9]. However, no standard experiment exists to demonstrate this benefit. This project developed an experiment to attempt to simulate these conditions. The reported experiments suggest commercially-fabricated FeCrAl alloy C26M cladding would be able to survive much longer dryout duration than conventional zirconium-based cladding materials.

2. METHODOLOGY

In the severe accident test station (SATS) located at Oak Ridge National Laboratory (ORNL), a novel experiment has been developed to gauge accident tolerance of fuel cladding materials under simulated cyclic dryout conditions [9,10] by modifying the parameters of a standard loss of coolant accident (LOCA) experiment [11]. Approximately ~30 cm lengths of cladding material (containing zirconia filler rods meant to imitate the internal volume occupation and thermal mass of fuel) were internally pressurized with Ar and subjected to thermal cycling in a steam environment. The FeCrAl alloy C26M tubing [7] had a wall thickness of 0.47 ± 0.02 mm and an outer diameter of 9.52 ± 0.03 mm, and the Zirc-2 had a wall thickness of 0.38 ± 0.01 mm and an outer diameter of 10.23 ± 0.02 mm. Cladding and test environment heating was provided by a quartz lamp infrared furnace [11]. The time, temperature, and extent of cycling was controlled by the input of a centrally located front thermocouple into LabView, combined with user set parameters. For this experiment, temperature cycling occurred from 300 °C to 650 °C.

Cladding assemblies were internally pressurized with Ar to 1 MPa and monitored via transducer to ensure proper sealing. Immediately afterwards, internal pressure was raised to a target nominal value and the assembly is then subjected to thermal cycling in a pure steam environment at atmospheric pressure, contained within a ~50mm ID custom quartz reaction tube. Steam was generated using distilled water as described previously [11]. Maximum cladding heating rates, determined with four thermocouples spanning the middle ~10 cm of the cladding surface, were found vary between materials but generally ranged from 13-17 °C/s. The temperature gradient from the top to bottom thermocouple was generally less than 50°C. The location and maximum temperature at cycle peak were also material specific but were found to occur within the same ~5 cm span of cladding opposite the front thermocouple. Internal pressure was monitored with a transducer. Swift and dramatic loss of pressure signaled cladding burst, at which point thermal cycling was terminated. Each presently reported cyclic dryout experiment was set to run fifty-four cycles however in many cases cladding burst and failure occurred within the first twenty cycles.

The true hoop stress σ_0 across the cladding tube was determined using the thin tube approximation,

$$\sigma_0 = \Delta p \frac{r}{w} \qquad (1)$$

where Δp is the radial pressure difference across the tube, r is the tube radius and w the tube wall thickness. Reported values of hoop stress, in MPa, were either determined from maximum observed pressure (at temperature peak of cycles) when no failure occurred, or pressure reading at time of burst when cladding failure did occur.

3. RESULTS

The temperature and hoop stress profile for a C26M tube, that did not fail after 54 cycles, is shown in Fig 1. The highest temperature reading occurred on the back side, opposite the front control thermocouple, approximately 5 cm from the center of the cladding assembly.

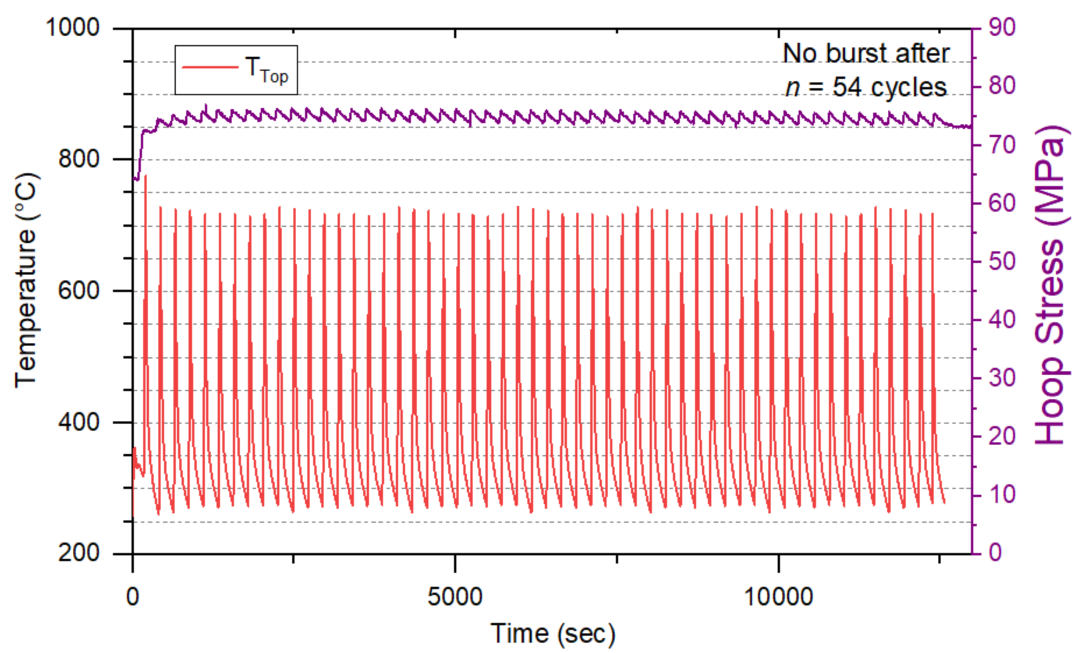

Figure 1: Temperature and hoop stress profile of a C26M tube during the simulated cyclic dryout experiment.

The temperature and hoop stress profile for a Zircaloy-2 tube, run with the same parameters as the C26M tube from Fig 1, is shown in Fig 2. The maximum mean temperature occurred on the back of the cladding assembly, opposite the front control thermocouple. The Zircaloy-2 tube failed during the fourth cycle.

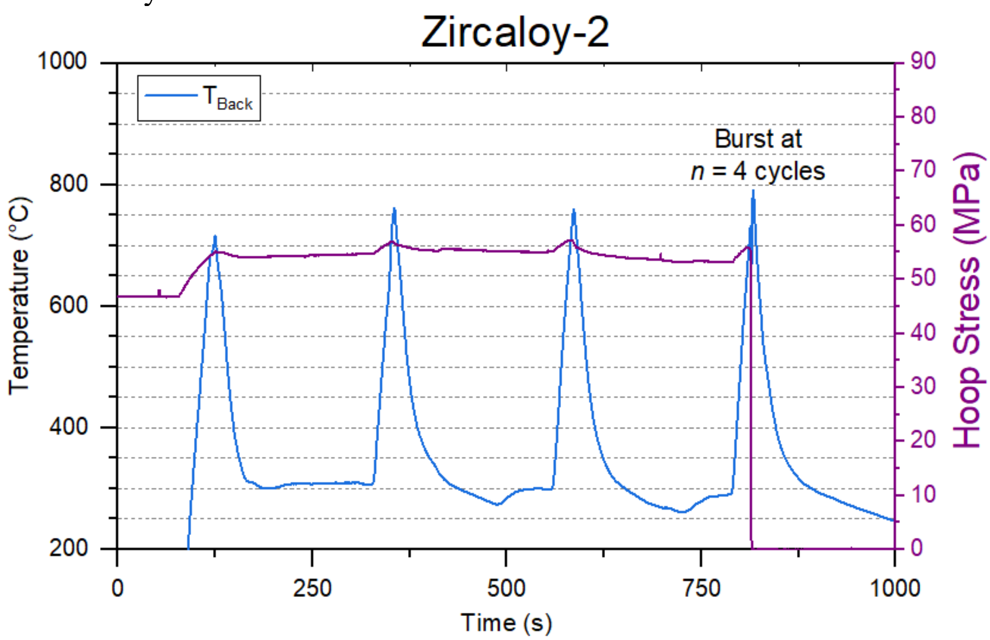

Figure 2: Temperature and hoop stress profile of a Zircaloy-2 tube during the simulated cyclic dryout experiment.

Time spent above temperature for the profiles in Fig 1 and 2, are shown in Fig 3. As can be seen, both tubes were subjected to similar thermal cycling, with both experiencing a maximum temperature of approximately ~780 °C. The hoop stress of the C26M tube was significantly higher than that of the Zircaloy-2 tube, 76 MPa at maximum peak temperature compared to 55 MPa at burst, respectively. The C26M tube is much more oxidation resistant than the Zircaloy-2 tube. Previously, a 1-2µm thick oxide was observed to form on Zircaloy-4 after 12 cycles to 650°C, while no oxide was observable on C26M [9]. In addition to oxide formation, internal oxidation (including both O and H ingress) of Zr-based cladding is expected to degrade performance and post-exposure characterization is in progress to determine the depth of ingress. At present based on the short time at temperature, another possible failure mechanism is that thermal cycling in steam may induce mechanical fatigue in the tube, eventually reaching a critical threshold at which the hoop stress induces tube burst. Additional experiments are needed at various cycle conditions and hoop stresses to fully appreciate the different responses.

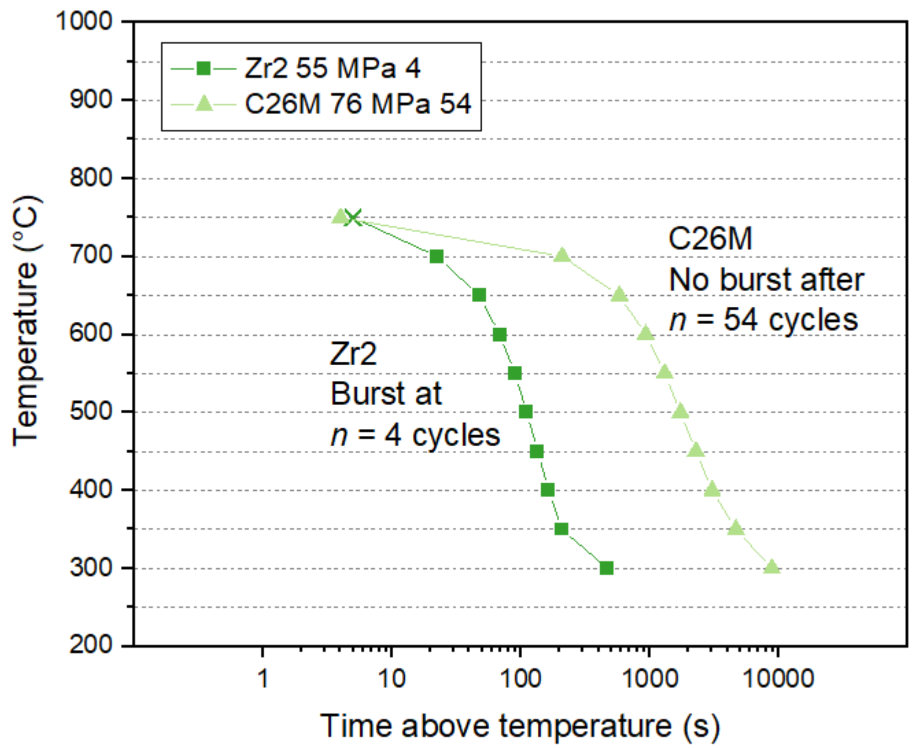

Figure 3: Dryout duration, or time spent above temperature, and respective temperature for the C26M and Zircaloy-2 tubes from Fig 1 and 2, respectively.

4. SUMMARY

A novel experiment to simulate cyclic dryout conditions has been developed. The results show an increased performance of FeCrAl alloy C26M over Zircaloy-2 under such conditions, an indication of C26M possessing superior high temperature mechanical properties and resistance to fatigue.

5. ACKNOWLEDGEMENTS

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6. REFERENCES

[1] HARA, T., MIZOKAMI, S., KUDO, Y., KOMURA, S., NAGATA, Y., MOROOKA, S. "Current status of the post Boiling transition research in Japan: Integrity evaluation of nuclear fuel assemblies after Boiling transition and development of rewetting correlations." *Journal of nuclear science and technology*, 40(10), 852-861 (2003).

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[2]. A. WYSOCKI, N. R. BROWN, K. A. TERRANI and D. M. WACHS, "Potential Impact of Cladding Wettability on LWR Transient Progression," *Trans. ANS* **115**, 473 (2016).
[3]. M. LIU, N. R. BROWN, K. A. TERRANI, A. F. ALI, E. D. BLANDFORD and D. M. WACHS, "Potential impact of accident tolerant fuel cladding critical heat flux characteristics on the high temperature phase of reactivity initiated accidents," *Annals of Nucl. Energy*, **110**, 48 (2017).
[4] Y. YAMAMOTO, B. A. PINT, K. A. TERRANI, K. G. Field, Y. Yang and L.L. Snead, "Development and Property Evaluation of Nuclear Grade Wrought FeCrAl Fuel Cladding for Light Water Reactors," *J. Nucl. Mater*., **467**, 703 (2015).
[5] B. A. PINT, K. A. TERRANI, M. P. BRADY, T. CHENG and J. R. KEISER, "High Temperature Oxidation of Fuel Cladding Candidate Materials in Steam-Hydrogen Environments," *J. Nucl. Mater*., **440**, 420 (2013
[6] B. A. Pint, "Performance of FeCrAl for Accident Tolerant Fuel Cladding in High Temperature Steam," Corrosion Reviews, 35(3) (2017) 167-175
[7] K. G. Field, Y. Yamamoto, B. A. Pint, M. N. Gussev and K. A. Terrani, "Accident Tolerant FeCrAl Fuel Cladding: Current Status Towards Commercialization, in Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, J.H. Jackson et al. (eds.), TMS, Warrendale, PA, 2018, pp.165-173.
[8] B. A. Pint, K. A. Terrani and R. B. Rebak, "Steam Oxidation Behavior of FeCrAl Cladding," in Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, J.H. Jackson et al. (eds.), TMS, Warrendale, PA, 2018, pp.235-244.
[9] B. A. Pint, L. A. Baldesberger, K. A. Kane, "Steam Oxidation, Burst and Critical Heat Flux Testing of Commercial FeCrAl Cladding," in Proceedings of the 19th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, J.H. Jackson et al. (eds.), ANS, 2020, in press.
[10] Pint, B. A. Steam Oxidation, Burst and Critical Heat Flux Testing in the Severe Accident Test Station (No. ORNL/LTR-2018/530). Oak Ridge National Lab.(ORNL), Oak Ridge, TN (2018).
[11] C. P. MASSEY, K. A. TERRANI, S. N. DRYEPONDT and B. A. PINT, "Cladding burst behavior of Fe-based alloys under LOCA," *J. Nucl. Mater*., **470**, 128 (2016).