Analysis of Cladding Property in Post LOCA Condition by Chromium Coating

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1. Introduction

After the Fukushima accident, the importance of nuclear power plant safety has been increased. In particular, attention was focused on the development of technology for the commercialization of Accident Tolerant Fuel (ATF). During the Loss of Coolant Accident (LOCA) scenario, a high temperature steam oxidation of the cladding occurred until the coolant system activated [1,2]. Combined with the pressure difference inner and outer cladding, ballooning and burst could happen.

In order to develop ATF, an assessment of accident resistance must be performed, which also led to an assessment of post-OCA material property. The main requirements of ATF are extending coping time and reducing the generation of hydrogen. Chromium has excellent abilities of high temperature oxidation resistance, high corrosion resistance, and reducing the generation of hydrogen [3]. For this reason, the chromium (Cr)-coated zirconium (Zr) alloy cladding is being considered as one of the candidates to replace the covering cladding in use currently.

In this research, the high temperature oxidation resistance of the previously developed (un-coated) cladding and the Cr-coated cladding that manufactured by two different conditions are quantitatively compared through the Post-quenched Ductility (PQD) test. The high temperature oxidation test was performed to simulate LOCA conditions. The change in condition results is compared with the ring compression test (RCT) as offset strain.

2. Test Methods and Results

2.1 Test Procedure

The current post LOCA acceptance criteria limit the Equivalent Cladding Reacted (ECR) with each hydrogen content level. Therefore, high temperature steam environment simulated tests were performed to get each ECR. RCT was performed to investigate oxidized cladding material properties.

LOCA experiments were conducted on both Cr-coated and uncoated cladding specimens to target ECR. Specimens were oxidized at 1204℃ to cover a range of target ECR and offset strains. The specimen holding zig allowed specimens to be located at the specified oxidation point [4,5]. After LOCA experiments, RCT was performed on the specimen at 135℃ and 0.033m/s [6,7]. This study complies with the guidelines described in the Regulatory Guide 1.223 of the U.S. Nuclear Regulatory Commission (NRC).

2.2 Material and Test Specimen

The Cr-coating was produced by an arc ion plating technique. The Cr was deposited on the outer surfaces of the cladding with an average 15μm thickness. In order to confirm the effect of two coating conditions selected for the production of Cr-coated cladding as shown in Table I. Test specimens were tested two-sided with Case 1 and Case 2-2 and one-sided with Case 2-1 and Case 2-2 oxidation experiments.

Table I: Specimen cases

Case	Coating	Coating procedure
Case 1	As received	-
Case 2-1	Coated	A
Case 2-2		B

2.3 PQD Test Results

As shown in Fig 1, it was confirmed that the Offset strain result of PQD tests were enhanced about 30% for the same ECR condition.

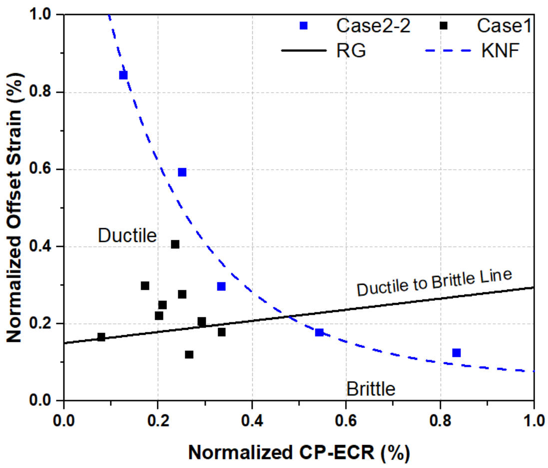

Fig. 1. Two-sided oxidation PQD test results

In Fig 2, through comparison of the coating procedure, it was confirmed whether there was no difference in results depending on the conditions.

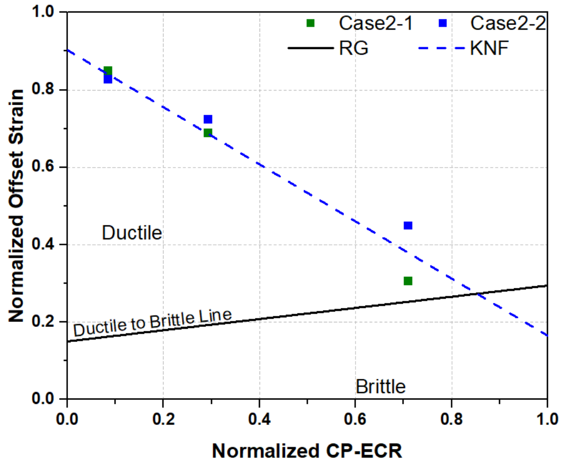

Fig. 2. One-sided oxidation PQD test results

3. Conclusions

In this study, the PQD test results according to the production process of the ATF Cr-coated cladding were compared into two categories.

First, the Cr-coated specimens show a performance enhancement of about 30% compared with un-coated specimens in the same ECR. The Cr coating reduces the oxidation occurring on the outer surface. The reason for enhancement in high temperature oxidation is that the Cr-O oxide layer has high temperature stability and protects the Zr alloy layer. At the PQD test, the Cr layer is a micrometer unit size that does not affect determining the material properties but affects the Zr alloy layer of the Cr-O oxide film that affects the material properties. Therefore, chromium coating prevents oxidation of Zr alloys and improves the PQD test results of the Cr-coated cladding.

Second, it was confirmed that there was no difference in the test results according to the difference in the coating procedure. The reason is that the Cr-coating layer throughs above the recrystallization temperature, and the difference in grains from the Cr-coating procedure disappears. Thus, the high temperature oxidation characteristics are not affected.

ACKNOWLEDGEMENT

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REFERENCES

[1] USNRC, 10 CFR part 50.46 Acceptance criteria for emergency core cooling systems for light water cooled nuclear power reactors, U.S. Federal Register, 2011.

[2] USNRC, 10 CFR part 50 Appendix K ECCS Evaluation Models, U.S. Federal Register, 2018.

[3] M. Shahin, J. Petrik, A. Seshadri, B. Phillips, and K. Shirvan, Experimental investigation of coldspray chromium cladding, Topfuel, pp. 1–10, 2018.

[4] M. Billone, Y. Yan, T. Burtseva, and R. Daum, Cladding Embrittlement During Postulated Loss-of-Coolant Accidents, NUREG/CR-6967, 2008.

[5] USNRC, Regulatory Guide 1.223 Determining post quench ductility, Washington. D.C., USA, 2018.

[6] K. Keum, Y. Lee, Effect of cooling rate on the residual ductility of Post-LOCA Zircaloy-4 cladding, Journal of Nuclear Material, Vol. 541, 2020.

[7] D.O. Hobson, P.L. Rittenhouse, Embrittlement of Zircaloy-Clad Fuel Rods by Steam during LOCA Transients, No. ORNL-4758, Oak Ridge National Lab., Tenn, 1972.