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Analysis of the Fuel-Clad Gap Region in High Burn-up PWR

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1. Introduction

Several recent studies have been conducted on fuel rod failures involving pellet-clad interaction (PCI)[1]. The results from these studies have established that PCI failures are due to stress corrosion cracking (SCC) of the cladding. A number of fission products (I, Cs, Cd, etc.) involved in the stress corrosion process have been identified. Of these, iodine and cadmium are prime suspects. Spent fuel rods are typically under tensile hoop stress because of the internal gas pressure in the rods. In typical PWR fuel rods, the end of life internal pressure can be significant because of the initial helium backfill pressure and the added pressure from the release of fission gases. Fuel-cladding interaction and the formation of fuel-cladding bonding layers with specified chemical, physical and mechanical properties are of importance with regard to the evolution of thermal conductivity as well as in the context of Pellet-Clad Mechanical Interaction (PCMI). It is also important in the framework of the long-term storage of spent fuel, where the phases formed at the fuel-cladding boundary are considered to be the first to be leached in the case of cladding failure [2].

2. Experimental preparations

A thin diamond wheel was cut off from a PWR fuel rod with a failed spent fuel rod with 53,000 MWd/tU and a normal spent fuel rod with 62,000 MWd/tU from a nuclear power plant withdrawn and cooled down for 2 and 4 years respectively. The samples have been embedded in epoxy resin and polished with diamond grinding disks of successively finer grain size, finishing on cloth with diamond paste of 1 ㎛ as a final stage. Before mounting the sample in the EPMA, the samples were coated with carbon to prevent charging. The carbon coated specimen was mounted in a holder together with the X-ray standard. The EPMA was performed on a JEOL 8230R equipped with a four wavelength dispersive X-ray spectrometer shielded with tungsten.

3. Results and Discussions

A photomicrograph of the clad-fuel gap section of the 62,000 MWd/tU fuel rod in the average oxygen region is presented in Fig.3, which also shows a quantitative analysis of O, Zr, and U on the fuel-clad gap in the 62,000 MWd/tU on the marked line, as shown in Fig. 3. The quantitative line scan data taken from across the fuel-clad gap shows stable states in an average oxygen region compared with the higher oxygen region, as presented in Fig. 4. At a series of radial positions, points quantitative analyses have been performed by covering the fuel-clad gap region to detect and measure any possible local concentration gradients of O, Zr, and U within an average oxygen region as shown in Fig. 3, and within a high oxygen region, as shown in Fig.4. Steps of 1 ㎛ were used to be able to assess the effective concentration gradients and resolution.

The concentration of O, Zr, and U illustrate the general trend in the fuel-clad gap regions as shown in Fig. 3, as compared to a high oxygen region as shown in Fig. 4. However, as shown in Fig. 4, the concentration distribution of Zr and U at the higher oxygen region is illustrating the general trend for redistribution in the fuel-clad gap region. These results indicate that even at a high burn-up state in the 62,000 MWd/tU, the fuel-clad gap reliability seems to be an appropriate condition

4. Conclusion

The concentrations of O, Zr, and U illustrate the general trend in the fuel-clad gap region, which is compared to a high oxygen region. However, the concentration distribution of Zr and U in the higher oxygen region illustrate the general trend of redistribution in the fuel-clad gap region. These results mean, that even at a high burn-up state in the 62,000 MWd/tU, the fuel-clad gap reliability seems to be an appropriate condition. The xenon and cesium concentration profiles illustrate the general trend for the release and cesium redistribution in the fuel-clad gap region in 84 ㎛ between clad to fuel. However, the oxygen and cerium concentration profiles illustrate a

remarkably different trend for the oxygen the fuel-clad gap region in 84 ㎛ between clad to fuel. The oxygen concentration profile at an average region is about less than 6 ㎛ compared to 20 ㎛ at the higher oxygen region between the clad to fuel gap. The quantitative line scan data of O, Zr, and U taken from across the fuel-clad gap show stable states in an oxygen-rich region, and the quantitative line scan data of fission products Cs, Ce, Ba, Sr, I show a general trend for release and redistribution in the fuel-clad gap region. These results indicate, that even at a high burn-up state at 62 GWd/tU, the fuel-clad gap reliability seems to be appropriate conditions.

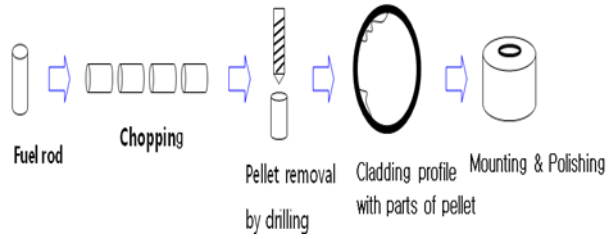

Fig.1 A thin diamond wheel was cut off from fuel.

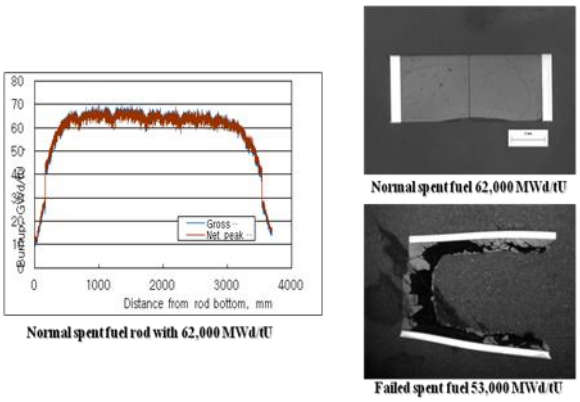

Fig.2 The failed spent fuel rod with 53,000 MWd/tU and normal spent fuel rod with 62,000 MWd/tU from nuclear power plant were discharged and cooled down for 2 and 4 years, respectively.

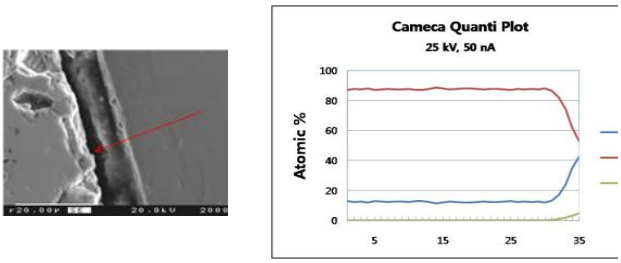

Fig.3 The SEM image and quantitative analysis of O, Zr, U within an average oxygen region. The concentrations of O, Zr, and U on the marked point illustrate the general trend for the release and redistribution in the fuel-clad gap region.

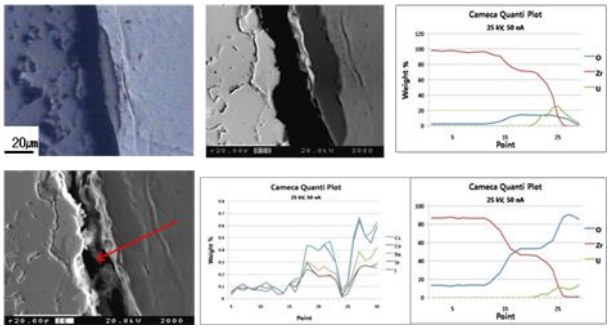

Fig. 4 Optical microscopy, SEM and BSE images of the 62,000 MWd/tU fuel rod in an oxygen content rich region and quantitative analyses of O, Zr, U on the marked point.

Reference

1. F.L. Yaggee, et al. “Characterization of irradiated Zircalloy: September 1989, EPRI NP-1155.
2. S.Van den berghe, A. Leenaers.., Seminar NEA proceedings on the PCI in Water Reactor Fuels, Aix-en-provance, France, 266, 9-11 March(2004)
3. H. Kleykamp, “The chemical state of the fission products in oxide fuels” J. Nuclear materials 131(1985)228.
4. H. Kleykamp, “The chemical state of the fission products in oxide fuels” J. Nuclear materials 131(1985)230.