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Fuel Integrity Evaluation for Ultrasonic Fuel Cleaning

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1. Introduction

KNFC is developing of ultrasonic device to remove CRUD deposits from the outer surface of PWR fuel rods. The CRUD removal device utilizes ultrasonic transducers to produce high frequency pressure pulses in the water surrounding the fuel assembly. As a consequence of the ultrasonic cleaning process, low frequency vibrations are induced in the cladding. This has prompted the fuel rod integrity issues such as pellet cracking and pulverization, cladding fatigue failure and PCI.

This paper summarizes the analysis performed to evaluate the fuel rod integrity related to ultrasonic CRUD cleaning of irradiated PWR fuel assemblies. The analysis utilized experimental data and observations from the full-size sectional mockup tests to perform fuel integrity evaluation through the finite element modeling of the fuel pellet and cladding response during cleaning.

2. Full Size Sectional Mock-up Tests

2.1 Full Size Sectional Mock-up Test

A prototype of fuel assembly cleaning device was fabricated to perform laboratory tests to determine its cleaning efficiency and to evaluate the impacts of cleaning on the fuel rod. The prototype device represents a full-size mockup of a single grid span location and the cleaning device. The mockup device includes four ultrasonic transducers, fuel assembly box tube, the vertical outer reflector and the structural side plates. A schematic of the full size sectional mockup is shown in Figure1.

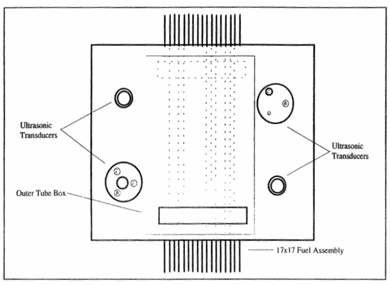

Figure 1. Full-Size Sectional Mockup Device

A series of test measurements using full-size sectional mockup device were conducted to obtain the fuel rod displacement and acceleration at various cell locations within the fuel assembly.

2.2 FE Analysis of the Full Size Sectional Mock-up Test

A finite element analysis was performed for full scale sectional mockup device to obtain a relationship between the tube free displacement and mid span displacement and derive an input acceleration forcing function that reproduces the observed cladding time history response.

The finite element model of sectional mockup consisted of a single cladding tube with rotational spring connections at the two grid spacer locations and is shown in Figure 2.

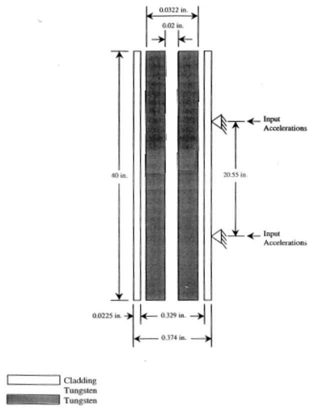

Figure 2. Finite Element Model of Single Rod

3. Fuel Pellet and Clad Integrity

The ultrasonic energy is transported through solid or liquid media, but gas regions act as ultrasonic energy insulators. Therefore the gas-filled gap between pellet and cladding should insulate the pellet from direct interaction with the ultrasonic energy. However the pellet-cladding gap is not uniformly distributed in fuel rod and some energy transfer through the cladding is possible. The direct energy deposition into pellet would result in cracking into small particles and these particles would locate into pellet-cladding gap and impinge onto the cladding, causing stress concentrations at the cladding inner surface during subsequent reactor startup of the fuel. Pellet-Cladding Interaction(PCI) failures in LWR fuel rod coincide with stress corrosion cracking at local stress concentrations in the cladding[2]. And the fuel rod vibration caused the cladding failure by fatigue depending on the cycles and the displacement amplitude. Pellet cracking and pulverization and cladding fatigue failure by ultrasonic energy should be evaluated to determine the potential for loss of fuel integrity.

3.1 Pellet Cracking and Pulverization

The lateral acceleration loads required to fracture a piece of fuel pellet can be calculated based on the diagram shown in Figure3.

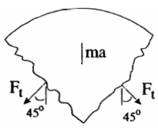

Figure 3. Schematic of Cracked Pellet Piece

By balancing the forces in Figure 3, the acceleration of fuel particle is given by;

$$a = \frac{2F_t \cos(\pi/4)}{m} \quad (1)$$

where, m is the mass of fuel pellet, F_t is the tensile force required to fracture the pellet. The tensile force can be written as;

$$F_t = \sigma_t \times A \quad (2)$$

where, σ_t is fracture stress for UO_2 fuel and A is fracture area of pellet.

Using equations of (1) and (2), the acceleration required to fracture the pellet was calculated. Compared to the measured cladding accelerations in full-size sectional mock-up test, this result is approximately a factor of 2 higher than any acceleration experience by the cladding.

3.2 Post Pellet-Cladding Interaction

The cracked pellet pieces could move onto the cladding due to fuel rod vibration during cleaning. This would result in PCI during subsequent reactor startup. To investigate the possibility of the motion of cracked pellet pieces during cleaning, a dynamic vibration analysis of full length fuel rod was performed using power density spectrum obtained from finite element analysis of full-size mockup sectional mockup test.

The dynamic analysis of full length fuel rod demonstrate that the maximum displacement of fuel rod during the cleaning is less than the flow induced vibration displacement of fuel rod in reactor during normal operation and also those of measured in the full-size sectional mockup test.

3.3 Cladding Fatigue Failure

Based on the displacement and frequency analysis results from the full size sectional mockup test, the cladding fatigue was evaluated using a simple beam analysis shown on Figure 4 and irradiated Zircaloy cladding fatigue failure curve[1].

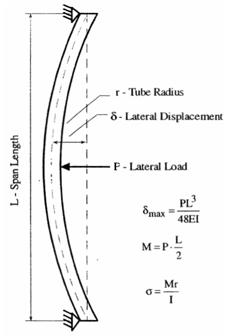

Figure 4. Simply-Supported Beam Diagram

The stress amplitude versus number of cycles is shown in Figure 5 and includes the bounding upper range during CRUD removal and fatigue design curve for irradiated Zircaloy. These results indicate a sufficient margin to fatigue failure for the vibrations experienced during the cleaning.

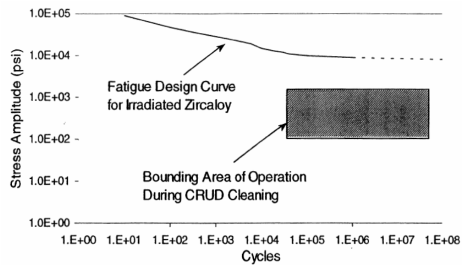

Figure 5. Stress Amplitude versus Number of Cycles

4. Conclusion

Based on the experimental observations from the full size sectional mockup test and analysis, the ultrasonic cleaning does not affect the fuel rod integrity.

REFERENCES

[1] O'Donnel and Langer, "Fatigue Design Bases for Zircaloy Components," Nuclear Science and Engineering, Volume 20, January 1964.

[2] ROSEbaum, H. S., et al., "Zirconium in the Nuclear Industry: Seventh International Symposium, ASTM STP939, 1987.