

## Pellet-Cladding Interaction of SNF under Postulated Drop Conditions

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### 1. Introduction

Spent nuclear fuel (SNF) contains numerous fission products and continuously emits radiation, posing potential risks to both human health and the environment. The SNF cladding serves as the primary barrier against the release of radioactive materials and its structural integrity must be ensured during transport [1].

In reactor operation, the fuel is exposed to high temperature, hydrogen precipitation, and neutron irradiation, which together induce pellet-cladding interaction (PCI) such as pellet expansion, cladding contraction, and the formation of a zirconia layer from oxidation [2]. The PCI substantially influences the long-term reliability of the cladding.

The finite element (FE) method is widely adopted to assess structural integrity of SNF [3]. In particular, researchers have developed numerical models that account for pellet expansion and the associated contraction of the surrounding tube to investigate its behavior [4]. Although such FE models provided useful insights into PCI, the zirconia layer also plays a significant role in nuclear fuel safety [5].

In this study, FE analyses were conducted to assess the effect of PCI on SNF cladding under postulated drop conditions. Two models were developed, one incorporating zirconia layer formation and the other without it. Three different drop heights were analyzed numerically, and the results were compared to determine the applicability of the models.

### 2. Analysis Method

#### 2.1 Analysis model

To focus on PCI effects, a partial model of the fuel assembly was constructed, as shown in Fig. 1. Two models were developed: zirconia consideration (ZC) and non-zirconia consideration (NZC). The former incorporated the oxide layer generated on the cladding under high burnup conditions, whereas the latter did not. Fig. 2 illustrates their cross-sectional details.

#### 2.2 Material properties

Material properties of Zircaloy-4 (cladding), UO<sub>2</sub> (pellet), and ZrO<sub>2</sub> (zirconia layer) were selected to reflect neutron irradiated condition, assuming 10 years of cooling [4]. The conditions were 250 °C, neutron fluence

of  $1.2 \times 10^{25}$  n/m<sup>2</sup>, burnup of 60 GWd/MTU, and hydrogen concentration of 600 wppm. The specific parameters are summarized in Table I.

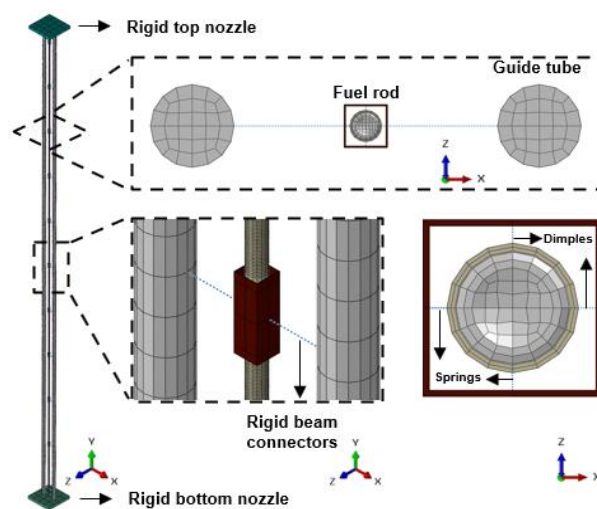


Fig. 1. FE model of a single fuel rod.

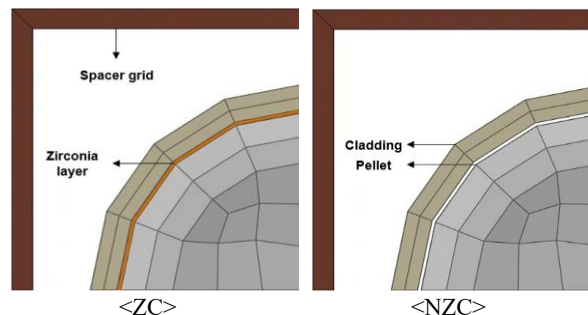


Fig. 2. Cross-sections of ZC and NZC models.

Table I: Material properties of fuel rod

Material	Density (kg/m <sup>3</sup> )	Elastic modulus (GPa)	Poisson's ratio
Zircaloy-4 [4]	6,590	77.64	0.328
UO <sub>2</sub> [4]	10,440	168.3	0.32
ZrO <sub>2</sub> [6]	5,769	143.98	0.3

#### 2.3 Analysis conditions

Drop heights of 3, 6, and 9 m were considered, with the corresponding rigid-body impact velocity calculated

using gravitational acceleration. In the ZC model, the cladding–zirconia interface was tied, while the zirconia–pellet interface was modeled with hard contact [7]. In the NZC model, an initial gap was defined between pellet and cladding, allowing contact to occur as deformation progressed. The fuel rod was modeled with free motion in all degrees of freedom, and the support effect of the spacer grid spring and dimple was implemented using linear spring elements.

### 3. Analysis Results

The analysis results are presented in Fig. 3. Under all drop conditions, the cladding exhibited the maximum principal strain in the lowest region between the bottom spacer grid and the adjacent grid. In addition, the ZC model showed significantly lower strain compared to the NZC model, and the reduction effect became more pronounced with increasing drop height. Specifically, the differences of strain were approximately 26% at 3 m, 60 % at 6 m, and 77% at 9 m. This indicates that zirconia mitigates PCI effects more effectively under severe impact conditions. The results of FE analyses are summarized in Table II.

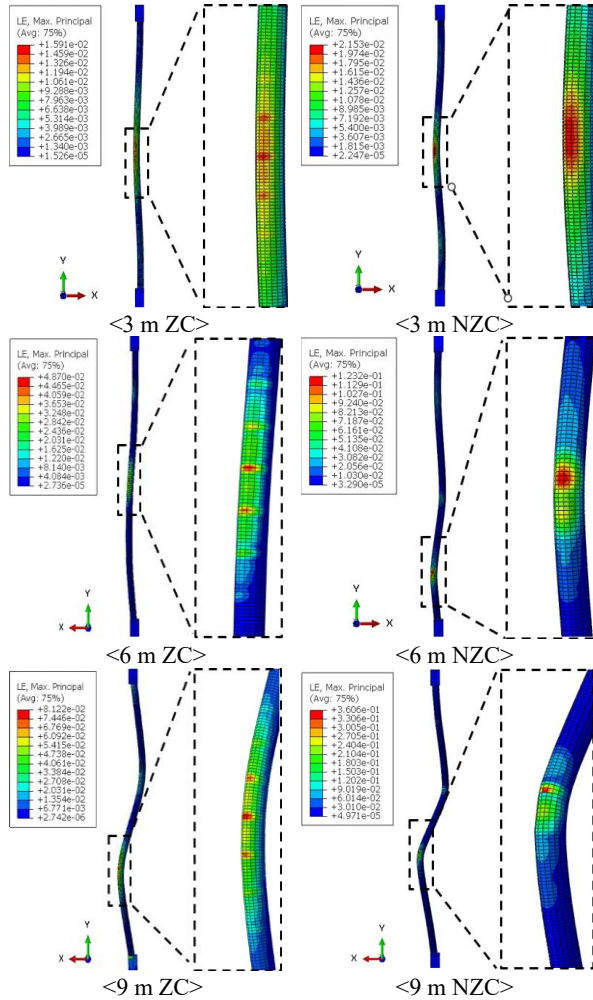


Fig. 3. Distributions of maximum principal strain in ZC and NZC models under drop conditions of 3 m, 6 m, and 9 m.

Table II: FE analysis results for maximum principal strain of cladding

Analysis condition	ZC (mm/mm)	NZC (mm/mm)	Difference (%)
3 m	1.59E-02	2.15E-02	26.04
6 m	4.87E-02	1.23E-01	60.41
9 m	8.12E-02	3.61E-01	77.51

### 4. Conclusions

In this study, FE analyses were performed under different heights to assess the effect of PCI on SNF cladding behavior. The main findings are summarized as follows:

- (1) In the zirconia layer was considered (ZC model), cladding strain was significantly reduced compared to the NZC model under all drop conditions.
- (2) As the drop height increased, PCI effects became more pronounced, and the zirconia layer exhibited a clearer mitigation effect.
- (3) The results assess that the proposed zirconia-considered model provides a basis for quantitatively assessing PCI effects, offering applicability for future SNF safety evaluations.

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### REFERENCES

- [1] U.S. Nuclear Regulatory Commission, Packaging and Transportation of Radioactive Material, Title 10, Code of Federal Regulation, Part 71, 1999.
- [2] X. Liu, M. N. Cinbiz, B. Kombaiah, L. He, F. Teng, E. Lacroix, Structure of the pellet-cladding interaction layer of a high-burnup Zr-Nb-O nuclear fuel cladding, *Journal of Nuclear Material*, Vol.556, 153196, 2021.
- [3] M. J. Park, Y. G. Shin, B. Almomani, Y. S. Chang, Comprehensive integrity assessment of spent nuclear fuel cladding during normal and postulated conditions of transportation, *Nuclear Engineering and Design*, Vol. 208, 105116, 2024.
- [4] B. Almomani, D. C. Jang, S. H. Lee, Structural integrity of a high-burnup spent fuel rod under drop impact considering pellet-clad interfacial bonding influence, *Nuclear Engineering and Design*, Vol.337, pp. 324-340, 2018.
- [5] N. Rodríguez-Villagra, L. J. Bonales, S. Fernández-Carretero, A. Milena-Pérez, L. Gutierrez, H. Galán, Exploring a surrogate of pellet–cladding interaction: Characterization and oxidation behavior, *MRS Advances* 8, pp. 238-242, 2023.
- [6] J. M. Cuta, S. R. Suffield, J. A. Fort, H. E. Adkins, Thermal Performance Sensitivity Studies in Support of Material Modeling for Extended Storage of Used Nuclear Fuel, U.S. Department of Energy, Used Fuel Disposition Campaign, FCRD-UFD-2013-000257, PNNL-22646, Sept. 27, 2013.
- [7] Dassault Systems, ABAQUS User's Manual, 2023.