Structural Integrity Evaluation of Spent Nuclear Fuel Assembly Under Normal Transportation Drop Conditions

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1. Abstracts

In this study, the structural integrity of the spent nuclear fuel assemblies was evaluated by carrying out a 0.3 m drop impact analysis, one of the normal transportation conditions of the nuclear fuel assemblies. For this purpose, the spent nuclear fuel assembly was modeled in detail as beam elements, and a coupled model for impact analysis was developed by inserting the modeled nuclear fuel assemblies into a cask.

2. Introduction

2.1 General information

Recently, the assessment of the integrity of spent fuel cladding for normal transport conditions as well as the hypothetical impact accident condition is becoming an increasingly important. The failure mode of the spent fuel cladding varies depending on the input conditions of the accident load, the shape of the fuel, and the material properties. It has been known that the elongation is known to be 1 ~ 4% and the maximum elongation varies with load and boundary conditions according to the results of physical properties of spent fuel cladding [1]. Therefore, in order to evaluate the integrity and failure mode of the cladding tube against for normal transport conditions as well as the hypothetical impact accident load, the influence of the cladding, the spacer grid, the tie plate and so on should be considered.

Sandia National Lab. suggested finite element modeling and analytical methodology for drop impact analysis of single spent fuel cladding and spent fuel assemblies, and describes the methodology for evaluating the integrity of spent fuel cladding [1].

USNRC and PNNL have described the finite element modeling method and the evaluation method using the shell and beam elements for the drop analysis of a single spent fuel cladding on the basis of the reference [1] [2].

In this study, the spent fuel assemblies were modeled in detail as beam elements [3], and a coupled model for impact analysis was developed by inserting a modeled fuel assembly finite element model into the cask.

The impact analysis of 0.3 m drop height, one of the normal transport conditions of the spent nuclear fuel assemblies, was performed and the integrity of the nuclear fuel assemblies was evaluated.

2.2 Analysis Conditions

The drop impact analysis in the normal transport condition was carried out with the finite element model including the fuel assemblies developed in Reference [3] as shown in Fig. 1, and the 0.3 m vertical and 0.3 m horizontal drop impact analysis were performed.

The following assumptions was made for the analysis of vertical and horizontal drop impacts.

- The spent fuel assemblies are CE type
- The drop height is 0.3 m
- The shock absorber is an isotropic material with a yield stress of 12.2 MPa
- Allowable failure strain of cladding is 0.04

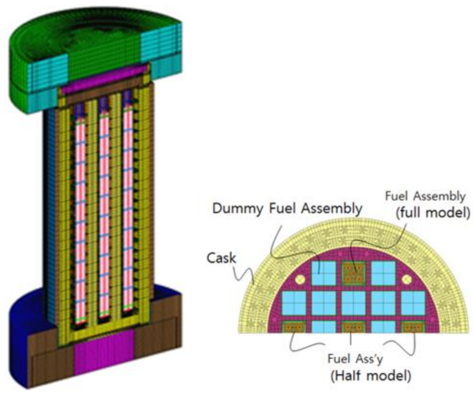

Fig. 1. Coupled Finite Element Model of Transport /Storage Cask and Spent Nuclear Fuel Assemblies.

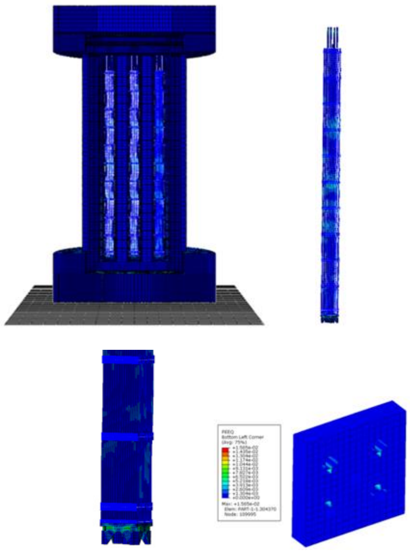

Fig. 2. Analysis Results for 0.3 m Vertical End Bottom Drop.

3. Analysis Results

3.1 0.3 m Vertical End Drop Analysis

Fig. 2 show the effective plastic strain distribution of the cask, the spent nuclear fuel assemblies and grid for the 0.3 m vertical end drop analysis.

No plastic deformation occurred in the cladding of the fuel assembly in the transport cask, and a small plastic strain of 0.018 was generated in the supporting grid

3.2 0.3 m Side Drop Analysis

No plastic deformation occurred in the cladding of the fuel assembly in the transport cask, but a relatively large plastic strain of 0.291 occurred in the supporting grid as shown in Fig. 3.

Table 1. Analysis Results for 0.3 m Drop Height in Normal Transportation Conditions

Drop Direction	Cask Acceleration G	Impact Limiter Deformation mm	Max Plastic Strain @ Cladding	Max Plastic Strain @ Grid
Bottom	40.67	11.69	0.000	0.016
Horizontal	22.32	23.55	0.000	0.291

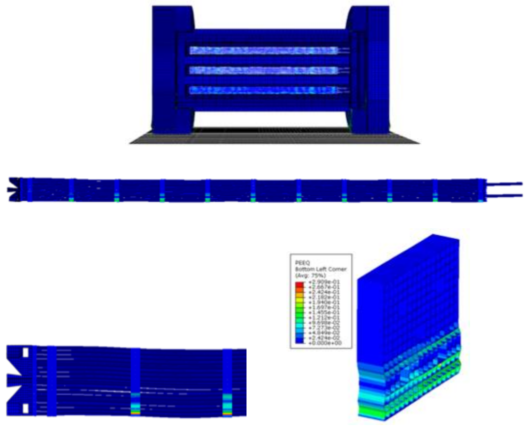

Fig. 3. Analysis Results for 0.3 m Horizontal Drop.

4. Conclusions

It was concluded that there is no problem in the integrity of the cladding of the nuclear fuel assemblies as well as the transport cask in the vertical and horizontal drop impact analyses at the normal transport condition of 0.3 m. However, in the case of a horizontal drop of 0.3 m under normal transport conditions, a relatively large plastic deformation or local failure may occur in some grids supporting the fuel claddings.

REFERENCES

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[3] S.S. Cho, W.S. Choi, K.S. Seo and Y.Y. Yang, "Drop Analysis of Spent Nuclear Fuel including Cladding", Proc. of the KRS 2016 Fall Conference, 14(2), Oct. 12-14, 2016, Jeju.