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Criticality Analysis for Storage Casks of Damaged Nuclear Fuel

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1. Introduction

Damaged nuclear fuel(DNF) is stored temporary at an interim site in the nuclear power plant and is expected to be deposited in the deep high level waste storage site in the future. In order to storing safely damaged nuclear fuel(DNF), it is suggested to be refabricated and to be stored in the specific cask by the existing dry fuel processes such as a DUPIC (Direct Use of PWR Spent Fuel in CANDU) process.[1][2] During the dry process of DNF, except some volatile isotopes most fission products and actinides are remained, thus it provides a good proliferation resistance. In this study, a transport and storage cask is proposed based on the existing 21 contained cask and the height is reduced to 2 m and 1 m for convenience of storing fuel assembly manufacturing from DNF. From the NUREG/CR-2216[4], under normal and abnormal conditions such as flooding with water moderator, the cask provides a sufficient criticality margin. Therefore, transport casks containing DUPIC fuel must be designed to remain subcritical under flooding conditions.

In this study, criticality safety was evaluated for transport cask models loaded with stabilized damaged fuel assemblies of 1 m, 2 m, and 4 m rod lengths, considering their structural adjustability. The design was based on specifications of domestically developed transport casks to enhance practicality and applicability [4]. The objective was to comply with NRC regulatory guidelines for transport and storage casks by maintaining the effective multiplication factor (k_{eff}) below 0.95 under flooded conditions, thereby determining the maximum allowable enrichment for each cask.

2. Design and Methodology

2.1 Design Specifications of Transport Cask

The transport cask designed in this study can accommodate 21 fuel assemblies, each consisting of 289 fuel rods. In conventional spent fuel assemblies, the guide tubes for instrumentation are left empty; however, in this study, these positions were utilized to insert additional fuel rods, thereby maximizing the storage capacity within the same volume as shown in Fig. 1 and Fig. 2.

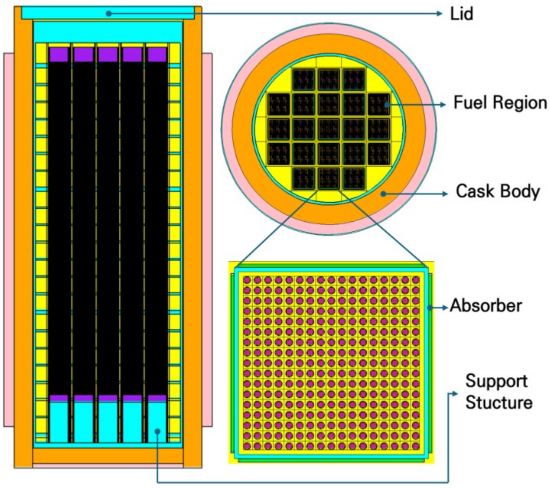
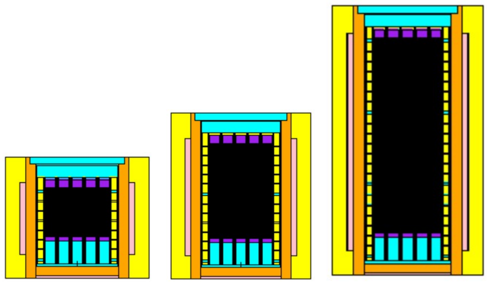


Fig. 1. Overview of transport cask(4m).



(a)Cask1(1m) (b)Cask2(2m) (c)Cask4(4m)

Fig. 2. Various casks for transportation of the DNF.

To evaluate the criticality characteristics according to fuel rod length, independent cask models were developed for 1 m, 2 m, and 4 m rods. For each case, the cask design was optimized in terms of internal support structures and impact absorption systems, with reference to models currently in use or under development in the Republic of Korea. The main design specifications of the transport cask, including overall dimensions and shielding thickness, are summarized in Table 1.

Table 1. Main Design Specification.

Items	Material	Dimension
Fuel	UO_2	D: 0.684 cm H: 1m / 2m / 4m
Cladding	Zircaloy-4	ID: 0.8cm OD: 0.8898cm

Neutron absorber	Boral	Thickness: 0.21cm Width:21.85cm
Lid & support structure	SS304	
Cask	Carbon Steel	ID: 169.6cm OD: 212.6cm

2.2 Computational Method

Criticality codes typically employ the Monte Carlo method to estimate the system k_{eff}. Among them, MCNP6.3[6] is chosen, which is developed by Los Alamos National Laboratory in the United States. And it is widely used not only criticality analysis but also radiation shielding analysis and high energy particle transport simulations. By utilizing the continuous-energy nuclear data library ENDF/B-VIII.0, the code avoids errors arising from group condensation, thus ensuring high computational accuracy. The convergence and uncertainty of Monte Carlo criticality calculations are sensitive to input parameters. In this study, simulations were performed with 10,000 neutrons per generation, 50 inactive generations, and a total of 3,100 generations.

3. Result

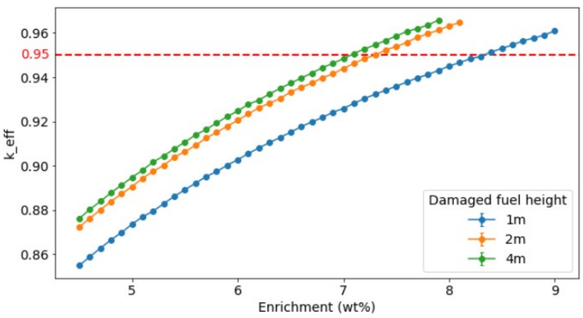


Fig. 3. Variation of k_{eff} with fuel enrichment for different damaged fuel lengths

Table 2. k_{eff} and Statistical Parameters for Different Damaged Fuel Lengths.

	Enrichment (wt%)	**k-effective**	**Standard Deviation (pcm)**
Cask1(1m)	(Max) 8.3	0.94943	15
	(Exceed) 8.4	0.95143	15
Cask2(2m)	(Max) 7.2	0.94846	15
	(Exceed) 7.3	0.94992	15
Cask4(4m)	(Max) 7.0	0.94837	15
	(Exceed) 7.1	0.95069	14

Fig.3 depicts the variation of k_{eff} with fuel enrichment for damaged fuel heights of 1 m, 2 m, and 4 m. Across all cases, k_{eff} increases with enrichment, with taller damaged fuel lengths because more fissile material is available for neutron multiplication. The statistical uncertainty for all Monte Carlo calculations, expressed in terms of one standard deviation, was maintained at or below 15pcm, which satisfies the high-precision Monte Carlo calculation quality criterion proposed by the OECD-NEA (≤ 50 pcm) and indicating a high level of statistical reliability in the results.[7] Based on the calculated values, the maximum enrichment that satisfies the subcriticality limit of k_{eff} <0.95 was found to be approximately 8.3 wt% for the 1 m case, 7.2 wt% for the 2 m case, and 7.0 wt% for the 4 m case. would cause k_{eff} to exceed 0.95 at the 95% confidence level, even when accounting for statistical uncertainty. These enrichment limits therefore represent the maximum allowable values to maintain the regulatory subcritical margin under the assumed flooded conditions.

3. Conclusion

In this study, the criticality safety of transport casks containing damaged nuclear fuel of varying lengths (1 m, 2 m, and 4 m) was evaluated using the MCNP6.3 code. The results showed that the statistical uncertainty (1σ) for all cases was maintained at or below 15 pcm, which is more stringent than the high-precision Monte Carlo calculation quality criterion (≤ 50 pcm) recommended by the OECD-NEA, thus ensuring statistical confidence in the results. For a given enrichment, longer damaged fuel lengths resulted in higher k_{eff} values. Under flooded conditions, the maximum enrichments satisfying the subcriticality limit of k_{eff} <0.95 were approximately 8.3 wt% for the 1 m case, 7.2 wt% for the 2 m case, and 7.0 wt% for the 4 m case. These findings provide design limits that meet both domestic and international regulatory guidelines and can serve as a basis for ensuring the safety of transporting and storing damaged spent nuclear fuel.

Acknowledgements

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