Preliminary Criticality Calculation for the Transport Cask of CE-Type 16×16 Spent Fuel Assemblies with the Overlapped-Type Neutron Absorber

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Spent nuclear fuels (SNF) involve fissile materials, fission products, and transuranium, so that they are highly radioactive and have a possibility to be critical state. Therefore, criticality safety for the transport cask loading spent fuel assemblies should be guaranteed through reliable evaluation procedure. According to the 10CFR71[1] and IAEA TS-R-1[2], the regulatory requirement for subcriticality is prescribed as follows: the k_{eff}, including all biases and uncertainties at a 95% confidence level, do not exceed 0.95. However, because there are uncertainties of modeling and possibility of design modification, the subcriticality limit is established that the maximum value of k_{eff} at a 95% confidence level do not exceed 0.935.

The objective of this study is to evaluate the criticality for the transport cask that is designed for transportation of 18, 21, and 24 fuel cells with the overlapped-type neutron absorber and to determine the minimum gap size between the adjacent fuel cells for each cask to meet the regulatory requirements. Figure 1 shows the typical outline and dimensions of fuel cell with overlapped-type neutron absorber.

Prior to criticality calculation, cavity (internal region of cask, exclusive of shields and the outermost structural material) inner radius must be determined with loading capacities and gap sizes between two fuel cells. Various errors, manufacturing tolerance included, has to be taken into account in the calculation process. For this reason, in this study, it is considered a margin of 2mm. As fuel cell type of the exclusive transport cask for CE-type 16×16 spent fuel assemblies is not yet definitely decided, the cavity inner radius must be determined as the maximum value out of calculation results for the two - overlapped and inserted - types. Table 1 contains these calculation results.

The method for performing the criticality analysis is the general three-dimensional Monte Carlo Code MCNPX v2.5.0. In this study, the reliability of computer code is verified by the modeling and executions of two critical experiments performed at Los Alamos Scientific Laboratory.[3][4] The assumptions used in 3-D modeling for criticality analyses are as follows.

- U-235 enrichment of all fuels: 5w/o
- Filling material in plenum and cavity: Water
- Setting up a water reflector cube 30cm thick on the outside of cask to consider the infinite arrangements
- Density of fuel pellets: 10.44g/㎤ (nominal value)
- Density of water: 1g/㎤ (Effective multiplication factor, k_{eff}, increases with water density.)
- Parts excluded from modeling of fuel assembly: Upper & lower end fitting, Spacer grids
- Parts excluded from modeling of cask body: Upper & lower trunnion, Heat transfer fins, Lid

For the criticality calculation using MCNPX code, it is required the data on density and isotopic composition of each material - fresh fuel (UO_2), cladding (zircaloy-4), neutron absorber (borated aluminum: E.P.1100), gamma shield (carbon steel: SA-350 Grade LF3), neutron shield (resin: NS-4-FR), supporting structure (stainless steel SA240 type 304&321), and water. Detailed information used for analysis is omitted in this paper.

And, in neutron problems, one neutron interaction table is required for each isotope or element in the problem. In this study, three continuous energy cross-section sets - ENDF/B-IV, ENDL92, and RMCCS - were used.

In this analysis, the KCODE card is used for calculating k_{eff}. This card specifies the MCNPX criticality source that is used for determining k_{eff}. The convergence of a Monte Carlo criticality problem is sensitive to the number of source histories per cycle, number of cycles to be skipped before averaging, and the total number of cycles to be done. Input values for each parameter are 10000, 10, and 100, respectively.

Table 2 presents the criticality results that are calculated with various loading capacities and gap sizes between fuel cells. As a result, considering the limit at the time of preliminary analysis, it is reasonable that the minimum gap sizes between fuel cells for cask to transport 18, 21, and 24 fuel cells are 20, 21, and 21mm, respectively.

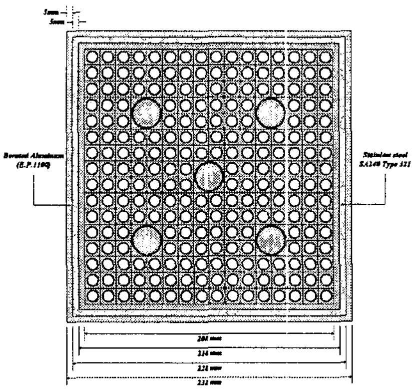

Figure 1. Schematic diagram of fuel cell with inserted-type neutron absorber

Table 1. Cavity inner radius for each loading capacity and gap size between fuel cells

Gap between fuel cells [mm]	Cavity inner radius for various loading capacities		
	18FA	21FA	24FA
17	713.6	736.9	804.8
18	715.8	739.1	807.3
19	717.9	741.3	809.8
20	720.0	743.6	812.4
21	722.1	745.8	814.9
22	724.2	748.0	817.4
23	726.4	750.2	820.0
24	728.5	752.5	822.5
25	730.6	754.7	825.1

Table 2. Criticality results for each loading capacity and gap size between fuel cells

Gap between fuel cells [mm]	Effective multiplication factor (k_{eff})					
	18FA		21FA		24FA	
	Mean	95% CI*	Mean	95% CI*	Mean	95% CI*
18	0.93760	0.93579 ~ 0.93941	-	-	-	-
19	0.93351	0.93172 ~ 0.93530	0.94165	0.93990 ~ 0.94339	-	-
20	0.92772	0.92587 ~ 0.92957	0.93524	0.93345 ~ 0.93704	0.93767	0.93595 ~ 0.93940
21	-	-	0.92956	0.92766 ~ 0.93146	0.93151	0.92947 ~ 0.93356
22	-	-	-	-	0.92646	0.92464 ~ 0.92828

*CI: Confidence interval

References

1. US Code of Federal Regulations 10CFR71, Packaging and Transportation of Radioactive Material, 1996.
2. IAEA Safety Standards Series No. TS-R-1, Regulations for the Safe Transport of Radioactive Material, 2005.
3. LA-UR-94-3270, Benchmark Critical Experiments of ^{233}U Spheres Surrounded by ^{235}U, 1995.
4. LA-UR-94-3275, Benchmark Critical Experiment of a Thorium Reflected Plutonium Sphere, 1995.