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Neutron and Gamma Shielding Evaluation for KN-12 Spent Nuclear Fuel Transport Cask

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1. Introduction

The CASTOR® KN-12 is designed to transport 12 intact PWR spent fuel assemblies for dry and wet transportation conditions. The overall cask length is 480.1 cm with a wall thickness 37.5 cm. Shield for the KN-12 is maintained by the thick walled cask body and the lid. For neutron shielding, polyethylene rods (PE) are arranged in longitudinal boreholes in the vessel wall and PE-plates are inserted between the cask lid and lid side shock absorber and between the cask bottom and bottom steel plate.

The shielding evaluation of the cask has been performed with MCNP to confirm the shielding integrity of cask for pre-service inspection of transport cask.

2. Calculations and Results

The spent fuel source terms were calculated by Origen-Arp [1] and Shielding calculations were made with MCNP5 [2]. The cask was modeled with its real dimensions, materials with composition and densities being taken from the KN-12 safety analysis report [3].

2.1 Calculations and Measurements

ORIGEN-ARP was run for each individual fuel assembly using the data given in table 1 for spent fuel having a burnup of ~ 31 GWD/MTU and an initial enrichment of 3.4 wt%. From these 12 runs, neutron and gamma source term for each assembly were calculated. The neutron and photon release for the 12 assemblies are shown in table 2.

Figure 1 shows a radial cross-sectional view of the modeled cask and spent fuels. The fuel assemblies are modeled to be homogenized over the physical cross section of the assembly. Axial cross-section of the model of the cask without impact limiters are also shown in figure 1.

All measurements were carried out with an isolated cask in spent fuel decontamination pit at KORI Nuclear Power Plant and Discrete measurements were performed at the 0° axis, 90 °, 180 °, and 270 ° for getting dose rate profiles.

Neutron dose rate measurements were performed with the commercial device PTS200 from Atlan Tech and the gammas dose rates were measured with the SURVEYOR 2000 from BICRON.

The shielding evaluation was performed for the cask with 12 particular fuel assemblies as described in table 1. The MCNP5 code was used for all the shielding analysis and a uniformly distributed Co-60 gamma source was assumed for the plenum, top nozzle, and bottom nozzle. Flux to dose conversion factor used in these calculations was taken from ICRP-74 [4].

Table 1 Description of Fuel Loaded into Cask

No.	Fuel ID	U235 Enrichment	Initial Uranium Mass (g)	Burnup (MWD /MTU)
1	G12	3.40	395,834	31,425
2	G29	3.40	396,006	31,425
3	G28	3.41	395,427	31,495
4	G06	3.40	396,341	31,844
5	G27	3.40	397,147	31,873
6	G03	3.40	396,264	31,598
7	G11	3.40	395,504	31,783
8	G31	3.38	396,377	31,923
9	G23	3.39	396,907	31,897
10	G08	3.40	396,012	31,753
11	G09	3.41	397,696	31,671
12	G22	3.41	398,562	31,617

Table 2. Neutron and Photon Release (#/sec)
(Basis: 410 kg U-metal, Enrichment: 3.4 % U-235. Burnup: 31,000 GWD/MTU, Assembly: 16 × 16, Cooling time: 17 yrs.)

	Fuel	Fuel Hardware	Top Nozzle	Plenum	Bottom Nozzle
Neutron	5.39e+7				
Photon	1.68e+15	1.84e+13	3.11e+11	4.75e+11	9.65e+11

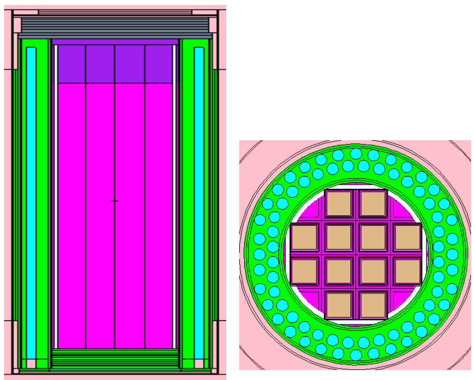

Figure 1. Cross sections of computational cask model.

2.2 Neutron Dose Rates

Figure 2 shows the calculated and measured dose rates profiles for the 0^{o} axis, 90 ^{o}, 180 ^{o}, and 270 ^{o} at axial cask surface. The C/E ratio of the axial neutron dose rates is around 1.0 ~ 2.56 except the trunnion region. Dose rates calculated for the trunnion region show the conservative results, because the trunnions are not modeled in this model. Dose rates profiles for top and bottom of cask are described in figure 3. Several points at top lid and bottom lid are compared between calculated and measured values. It is shown that the C/E ratios at top lid region are some higher than others in figure 3.

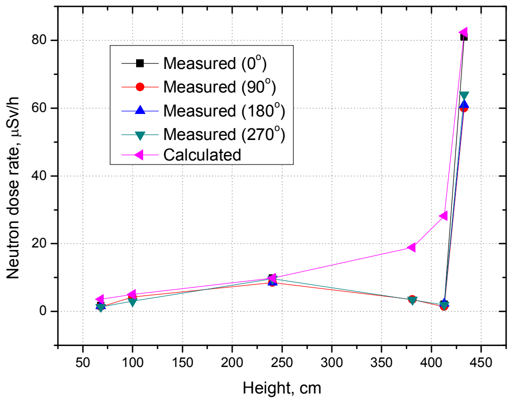

Figure 2. Axial neutron dose rate profiles.

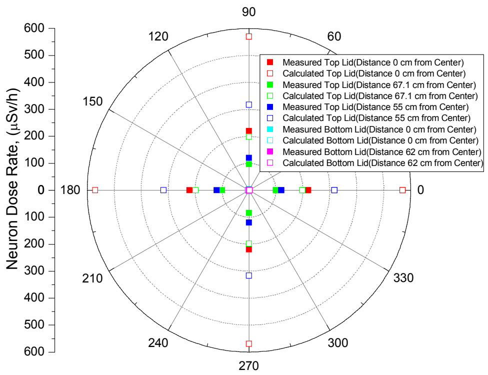

Figure 3. Circumferential neutron dose rate profiles at the top and bottom of cask.

2.3 Gamma Dose Rates

Figure 4 shows the calculated and measured dose rates profiles for the 0^{o} axis, 90^{o}, 180^{o}, and 270^{o}. The C/E ratio at center of the cask side is 3.0 at the 0^{o} axis, 90 ^{o}, 270 ^{o} and 2.3 at the 180 ^{o}. Figure 5 illustrates circumferential gamma dose rate profiles at top and bottom lid of cask. The C/E ratio is 1.1 ~ 1.8 at bottom regions, and 1.0 ~ 2.2 at top of the cask.

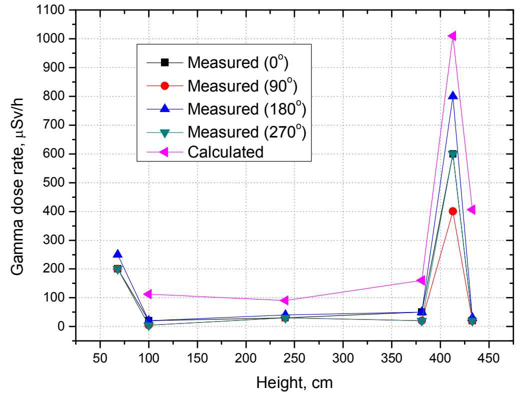

Figure 4. Axial gamma dose rate profiles.

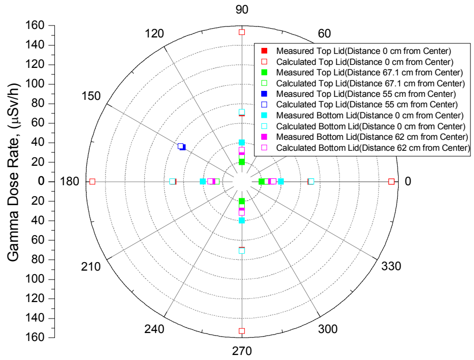

Figure 5. Circumferential gamma dose rate profiles at the top and bottom of cask.

3. Conclusion

The comparison between the calculated and measured dose rate shows good agreement for all axial, top and bottom profiles, i.e. the same principle trends with an overestimation of the measured dose rate by the calculations The C/E ratios are generally around 1.0 ~ 2.56 for neutron dose rates and 3.0 for gamma dose rates at center of the cask side and 1.0 ~ 2.2 at top and bottom lid based on the measurements data.

REFERENCES

[1]"SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations", ORNL/TM-2005/39, Version 5.1, November 2006.

[2] X-5 Monte Carlo Team, "MCNP: A General Monte Carlo N-Particle Transport Code, Version 5", LA-UR-03-1987, Los Alamos National Laboratory, April 2003.

[3]"Castor KN-12 Spent Nuclear Fuel Transport Cask-Safety Analysis Report," GNB B 002/2002.

[4]"Conversion Coefficients for use in Radiological Protection against External Radiation," Annals of the ICRP, Publication 74, ISSN 0146-6453, 1996.