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SHIELDING INTEGRITY TEST FOR PRE-SERVICE INSPECTION OF KN-12 TRANSPORT CASK

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ABSTRACT

The calculated and measured shielding evaluation of the cask for 12 spent nuclear fuels has been performed to confirm the shielding integrity of cask for pre-service inspection of KN-12 transport cask. The spent fuel source terms were calculated by Origen-Arp and Shielding calculations were made with MCNP5. The shielding evaluation was performed for the cask with 12 particular fuel assemblies and a uniformly distributed Co-60 gamma source was assumed for the plenum, top nozzle, and bottom nozzle. All measurements were carried out with an isolated cask in spent fuel decontamination pit at KORI Nuclear Power Plant in Korea and discrete measurements were performed at the 0^{o} axis, 90^{o}, 180^{o}, and 270^{o} 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. By comparison of calculated neutron dose rates with 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. Several points at top lid and bottom lid are also compared between calculated and measured values. It is shown that the C/E ratios at top lid region are some higher than others. Gamma dose rates were also compared each other with same procedure of neutron case. 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}. The result shows that C/E ratio is 2.1 ~ 3.4 at bottom regions, and 1.0 ~ 1.9 at top of the cask. 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.

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 of

37.5 cm. The maximum spent fuel assembly specifications of the cask were less than 50,000 MWD/tU of an average burn-up basis with 5.0 wt% enrichments and more than a 7 year cooling time. Shield for the KN 12 is maintained by a thick walled cask body and lid. For a 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 the bottom steel plate.

A schematic diagram of the KN-12 transport cask is shown in **Figure 1**. The cylindrical cask cavity has an internal diameter of 1,192 mm and an internal length of 4,190 mm. The lid is 290 mm thick and each impact limiter is 2,450 mm in diameter and extends 700 mm along the side of the cask in the axial direction[1)]. For a neutron shielding, polyethylene rods (PE) were arranged in the longitudinal boreholes in the vessel wall and PE plates were inserted between the cask lid and the lid side shock absorber and between the cask bottom and the bottom steel plate.

In February 2007, several tests for an inspection of its usage were completed to reuse the KN-12 cask under the provisions of the text of the Article 90-3 of the Korea Atomic Energy Act[2)]. A nuclear power-related operator who wishes to continue to use transport containers shall undergo an inspection of use with a 5 year time period. Because the KN-12 cask had been manufactured with a license in accordance with Korea Atomic Energy Act for Type B(U)F in 2002, a test program for an inspection of its usage was fulfilled in this year. The shielding evaluation of the cask has been performed with MCNP5[3)] to confirm the shielding integrity of the cask for a pre-service inspection of a transport cask. The test items for an inspection of its usage included a visual inspection, nondestructive weld inspection, load test, maximum operating pressure test, leakage test, heat transfer performance test, external surface contamination test and a shielding performance test, etc. The C/E ratios of the dose rates for the KN-12 transport cask were compared to check on the shielding performance of the KN-12 cask.

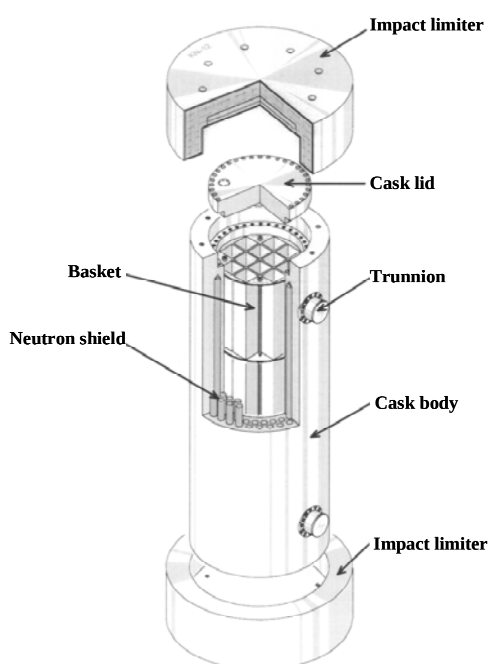


Figure 1. A schematic view of the KN-12 transport cask

THE CALCULATIONS AND MEASUREMENTS

Neutron and gamma source inventories for each assembly were calculated by the ORIGEN-ARP module in SCALE5[4)] with its cooling time to obtain the neutron and photon release rates of the loading assemblies. The spent fuel source terms were calculated by Origen-Arp and the shielding calculations were made with MCNP5. The cask was modeled with its real dimensions and materials with its composition and densities taken from the KN-12 safety analysis report[5)].

Source term

Source inventory was calculated with each individual fuel assembly of an initial enrichment of 3.4 wt% with average burn-ups of around 31 GWD/tU and 33 GWD/tU and 17 and 16 years cooling time. The typical calculated neutron and photon releases of the fuel assembly, e.g., are shown in **Table 1**. The assembly was divided as 5 sub-parts to calculate the source terms. The five sub-parts consisted of fuel parts, fuel hardware, plenum, and upper bottom plenum regions. **Figure 2** shows the calculated total neutron and gamma release rates. Neutron release rate was 6.73×10^{8} n/sec and the photon release rate was 2.06×10^{16} ɣ/sec up to a 10 MeV particle energy, respectively. The energy bin for the shielding calculation was 18 groups for the photon and 44 groups for the neutron.

Table 1. Neutron and Photon Release (#/sec) (Basis: 400 kgU, Enrichment: 3.4 % U-235. Burnup: 33,000 GWD/MTU, Assembly: 16 by 16, Cooling time: 16 yrs)

	Fuel	Fuel Hardware	Top Nozzle	Plenum	Bottom Nozzle
Neutron	5.85e+7				
Photon	1.71e+15	1.88e+13	3.07e+11	4.85e+11	9.86e+11

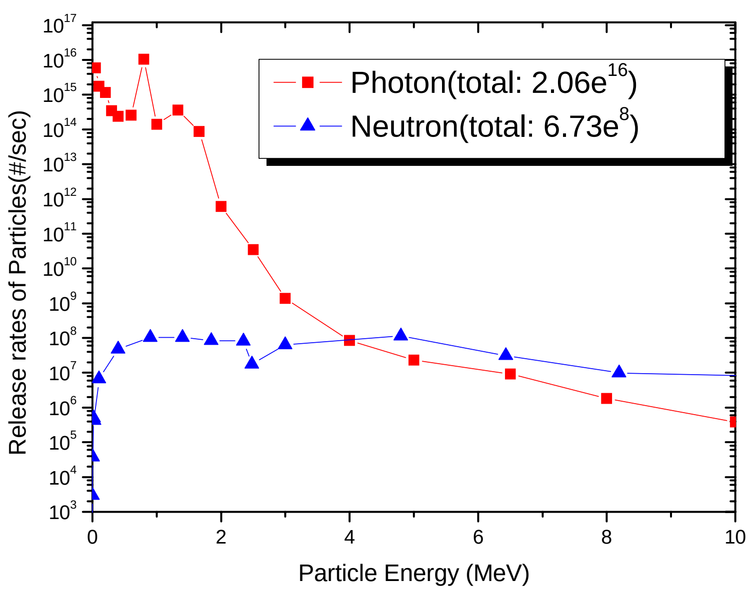


Figure 2. Total neutron and photon release rates of 12 spent fuel assemblies (basis: 400 kg U, Enrichment: 3.4 % U-235. Burnup: 31,000 ~ 33,000 MWD/TU, Assembly: 16 by 16, Cooling time: 16 ~ 17 years).

Shielding calculations

Axial and radial cross-sectional view of the modeled cask and the spent fuels is shown in **Figure 3**. The fuel assemblies were modeled to be homogenized over a physical cross section of the assembly. Above and below the fuel zones, i.e., plenum, top nozzle and bottom nozzle, were assumed to be a uniformly distributed ^{60}Co photon source. The fuel region was homogenized to consist of UO_2 and cladding. The cask body was modeled as SA-350 GRADE LF3(ASME) with a density of 7.8 g/cm^3 and the lid was modeled as SA-182 GRADE F6NM with a density of 7.89 g/cm^3. The density of the polyethylene homogenized in the moderator boreholes was assumed to be 0.87 g/cm^3. The trunnions were neglected in the cask shielding model. This assumption will provide more conservative results with regard to the measured results, because it provides an additional shielding to decrease the dose rates in the trunnions area. F2 surface tallies were used by placing the tally surfaces in several radial and axial locations in order to evaluate the dose rates around the cask. Flux to dose conversion factor used in these calculations was taken from ICRP-74[6)].

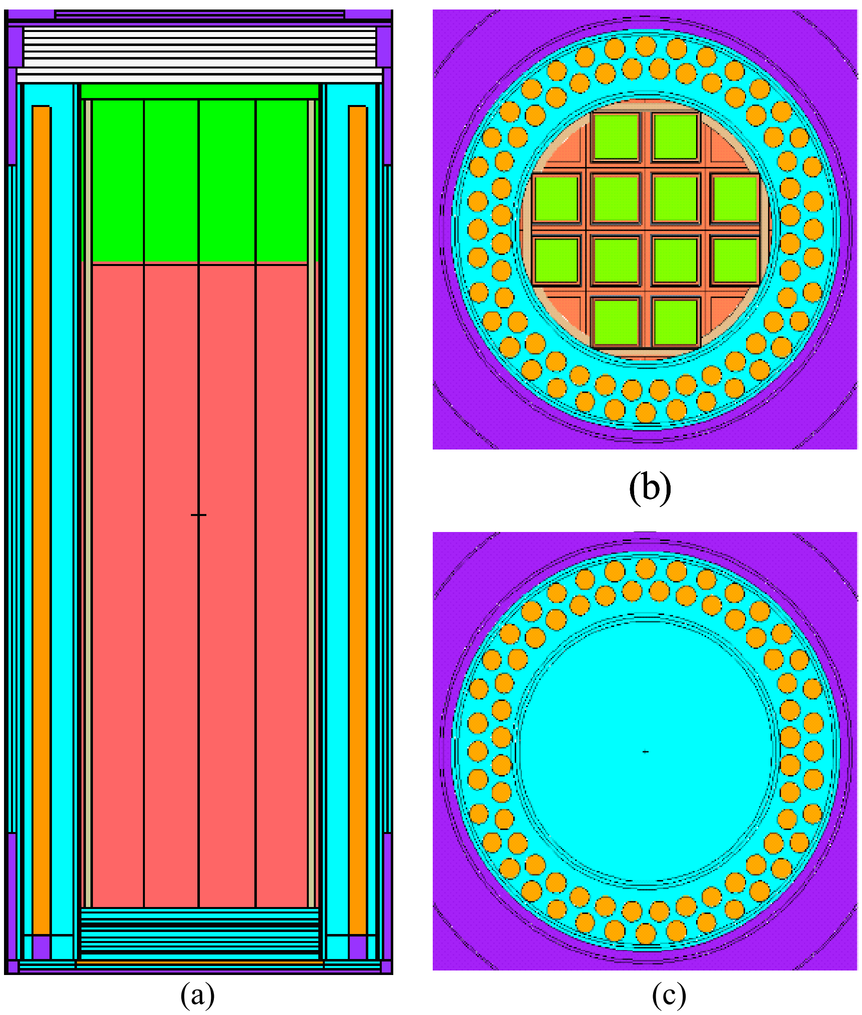


Figure 3. Cross sections of computational cask model. (a) axial MCNP model, (b) radial MCNP model at fuel region , and (c) radial MCNP model at bottom region.

Measurements

All the measurements were carried out with an isolated cask in a spent fuel decontamination pit at the KORI Nuclear Power Plant and discrete measurements were performed at a 0 °, 90 °, 180 °, and 270 ° axis to obtain dose rate profiles. **Figure 4** shows the test points of the KN-12 cask along the axial direction at 0 °, 90 °, 180 °, and 270 ° circumferentially and the top and bottom lid regions. **Figure 5** shows the measurement positions of several test points with detectors on the top lid, side surface, and bottom lid. All the points were carefully tested in accordance with the shielding integrity test procedures to protect an individual from an abnormal exposure. Neutron dose rates were measured with a commercial device PTS200 from Atlan Tech and the gammas dose rates were measured with the SURVEYOR 2000 from BICRON.

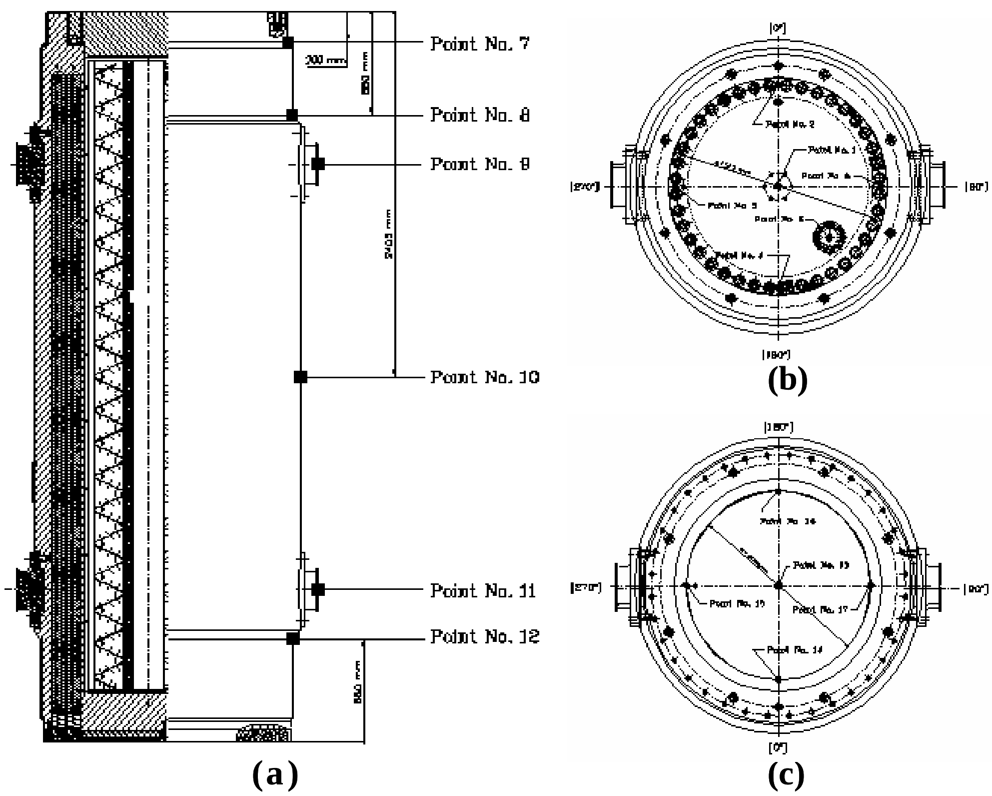
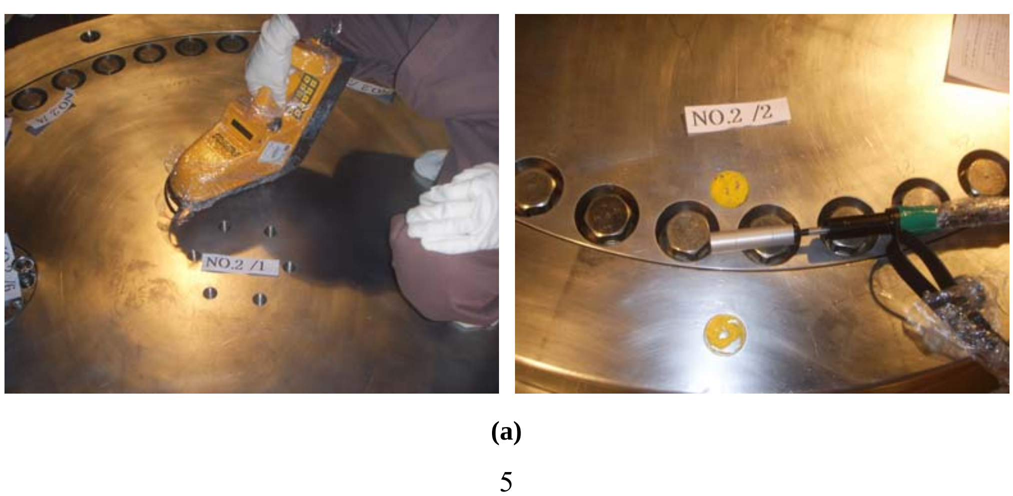


Figure 4. Dose rates test points of the KN-12 cask. (a) test points along the axial direction (b) points on top lid, and (c) points on bottom lid.



(a)

(b)

Figure 5. Gamma and neutron dose rates measurement points (a) on the top lid, (b) trunnions and the cask side surface.

RESULTS

Neutron Dose Rates

Figure 6 shows the calculated and measured dose rate profiles for the 0^{o}, 90^{o}, 180^{o}, and 270^{o} axis at the axial cask surface. The C/E values show a good agreement for all the axial and radial profiles with the trend of an overestimation of the measured dose rate by the calculations. The calculated highest neutron dose rate was below 100 μSv/h and this value is well agreed with measured values. The C/E ratio of the axial neutron dose rates is around 1.0 ~ 2.56 except for the trunnion region. Although the measured dose rates at trunnion showed the lowest values, the calculation results showed some deviation from measured values. Dose rates calculated for the trunnion region show conservative results, because the trunnion regions are not modeled in this model. And it would also stem from a bad modeling of the inside of the cask at the upper parts by the complex geometry of the assembly, source conditions, and an unsuitable tally surface division. The other points showed good agreements between the measured and calculated values.

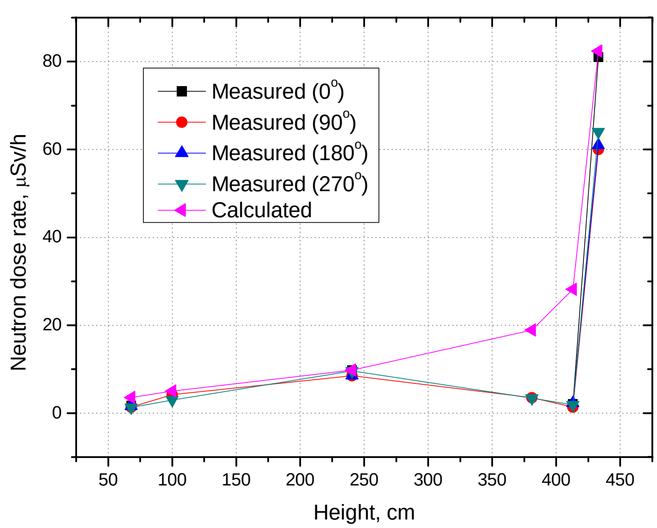


Figure 6. Axial neutron dose rate profiles.

Dose rates profiles for the top and bottom of the cask are shown in **Table 2**. Several points at the top lid and bottom lid are compared between the calculated and measured values. Results at the top lid showed a consistency between measured and calculated values, but C/E ratios at the bottom lid are somewhat higher than the others in Figure 6.

Table 2. Measured and Calculated neutron dose rates at the top and bottom lid of the cask

No.	Top lid Measured (μSv/h)	Top lid Calculated (μSv/h)	No.	Bottom lid Measured (μSv/h)	Bottom lid Calculated (μSv/h)
1	220	627.2	13	0.6	4.4
2	100	217.5	14	0.3	2.9
3	85	217.5	15	0.3	2.9
4	100	217.5	16	0.4	2.9
5	97	217.5	17	0.3	2.9
6	120	349.2			

Gamma Dose Rates

Figure 7 shows the gamma calculated and measured dose rate profiles for the 0 ^{o}, 90 ^{o}, 180 ^{o}, and 270 ^{o} axis at the axial cask surface. The C/E ratio at the center of the cask side is 3.0 at the 0 ^{o}, 90 ^{o}, 270 ^{o} axis and 2.3 at the 180 ^{o} axis. The highest gamma dose rate at cask side surface is 841.8 μSv/h by the calculation and 800 μSv/h by measurements. The measured dose rates at the lower and upper trunnions are 10 μSv/h and 20 ~ 50 μSv/h.

The measured dose rates at bottom side surface are around 200 μSv/h, which is some lower than calculated dose rates of 259.5 μSv/h. Dose rates on the top and bottom lid of the cask are shown in **Table 3**. The C/E ratio is 2.1 ~ 3.4 at the bottom regions, and 1.1 ~ 1.9 at the top of the cask. The gamma dose rates on the top show lower values than the neutron cases.

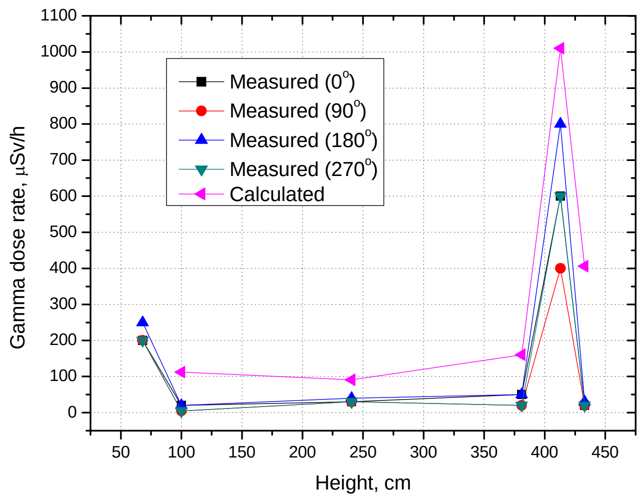


Figure 7. Axial gamma dose rate profiles.

Table 3. Measured and Calculated gamma dose rates at the top and bottom lid of the cask

No.	Top lid Measured (μSv/h)	Top lid Calculated (μSv/h)	No.	Bottom lid Measured (μSv/h)	Bottom lid Calculated (μSv/h)
1	70	97.7	13	40	136.6
2	20	37.9	14	30	63.1
3	20	37.9	15	30	63.1
4	30	37.9	16	30	63.1
5	20	37.9	17	30	63.1
6	70	75.2			

CONCLUSIONS

The C/E ratios for the neutron cases were generally around 1.0 ~ 2.6. The C/E ratio was 2.3 ~ 3.0 for the gamma dose rates at the center of the cask's side. The dose rates at top lid of the cask and bottom side show the far less values than those of cask side. The comparison between the calculated and measured dose rates showed a good agreement for all the axial, top and bottom profiles, i.e. the same principle trends with an overestimation of the measured dose rate by the calculations.

By this program the shielding integrity test for pre-service inspection of KN-12 transport cask was successfully completed by the neutron and gamma dose rates calculation and measurement. It revealed that all calculated and measured values showed the less values than 2000 μSv/h, which is the legal criteria of transport SF.

But the C/E ratios at the top lid of the cask revealed some discrepancy in this modeling, a more precise modeling at the top lid would be required.

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