Preliminary Analysis of Radiation Shielding for B-type HIC Transport Package

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1. Introduction

A radiation shielding analysis has been conducted using a computer program MCNP5 for a B-type HIC (High Integrated Container) Transport Package, which contains HIC with radioactive waste or Spent Resin, for transportation from nuclear power plant sites to disposal repository. Radiation source term is first carefully determined from the safety analysis reports related to HIC for appropriate calculation. And then MCNP (v.5) is performed to obtain the minimum thickness of HIC transport package, which meets the dose rate limit for HIC transport package prescribed in Korea Nuclear Law and IAEA Safety Standards for Radioactive Material Transport. In addition, some other analyses are done about the trend of dose rates depending on the thickness of shielding material and distance from the package.

2. Regulations

There are several regulations related to radioactive waste transport in the domestic and international laws. Focused on the objective of this article, the related regulations about radiation dose rate limit for radioactive waste transport package (including B-type transport package) is in Table 1. Dose rate at the surface of the package is limited to 2mSv/h and one at any point 2m away from the package is limited to 0.1mSv/h. The detection position could be on the radial direction (side) or axial direction (upside) of the package, and one which has relatively higher value than the other will be selected for the comparison with dose rate limit.

Category	Dose rate limit [mSv/h]
Surface of the package	2
2m away from the package	0.1

Table 1. Radiation dose rate limit for radioactive material transport package

3. Calculations and Results

In this section, methods to calculation and conditions required to perform a calculation are described.

3.1 Method to analysis

MCNP is one of the most useful codes performing criticality and shielding calculations using Monte Carlo Method. Before performing MCNP calculation, followings are required to get appropriate results: geometry of a HIC transport package, material properties of structures, radiation source, and detector positions. And because of the thickness of the steel shielding wall of HIC transport package, which is thicker than several centimeters, geometrical splitting method is used for MCNP shielding calculation.

After MCNP calculations with trial-and-error way, the minimum thicknesses of radiation shielding walls satisfying the regulation limit can be obtained and these are the objectives of the calculation.

3.2 Geometry and material

The schematic geometry of HIC transport package for MCNP calculation is in Figure1. It is in cylindrical shape and has a lid on the top. This cask can carry a HIC or spent resin in it. There are two impact limiters on the bottom and top of the cask, but in the calculation, these are eliminated for conservative result.

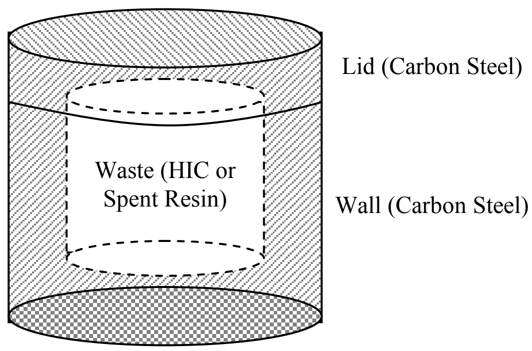

Figure 1. Schematic figure of HIC transport package

All metals used in the calculation are assumed to be carbon steel as shielding material for HIC transport package. The dry spent resin is used as a waste material and its average radiation dose rate is used for a dose rate of source term, and homogenized material composition is assumed for it. The compositions of these materials are shown in Table 2.

Material	Density [g/cm^3]	Composition [wt%]

Carbon steel	7.85	Fe(93.95) Mn(0.89) C(0.43) Cu(0.46) Si(0.15) S(0.01) P(0.02) Ni(3.94) Mo(0.21) V(0.03) Cr(0.28) Nb(0.03)
Dry spent resin	0.68	C(45.5) S(13.6) H(4.30) O(32.7) B(2.40) Fe(0.50) Na(1.00)

Table 2. Composition of materials for calculation

3.3 Source

The average radioactivity for each categorized isotope in dry spent resin generated from domestic nuclear power plants is referred in the calculation. Among many elements from the above data, several radioisotopes emit gamma, beta, and alpha rays, but contributions of beta and alpha rays to radiation dose rate are negligibly small compared to gamma ray, so the radioisotopes emitting gamma ray are considered as radiation sources in this calculation. And also gamma isotopes with very small amount or yield comparing with major gamma isotopes are neglected. Table 3 shows the radioisotopes with gamma ray used in the calculation.

Isotope	Energy [MeV]	Yield	Activity [Bq]
Co-58	0.811	0.995	1.30E12
Co-60	1.173	0.999	4.29E12
Co-60	1.332	0.999	4.29E12
Cs-137	0.661	0.661	3.18E12
Ce-144	0.134	0.134	4.24E09
Total			13.06E13

Table 3. Gamma ray source

3.4 Detector positions

In this MCNP calculation, virtual detectors (i.e. F5 tally card) are placed at considering points to obtain the information about dose rate. Two detectors are placed on the surface of the package and two other detectors are at the points 2m away from the surface. These detectors are placed usually on the center line of the package (side and upside) since the dose rate could be the largest in it. These virtual detectors in MCNP calculation simulate the real detectors on the transport package in the real situation. The counting values from virtual detectors are the values expected from the calculation.

3.5 Results

The calculation results are summarized in Figure 2 and Table 4. They show the minimum thicknesses of the lid, bottom and side shielding wall of HIC transport package within the regulatory limit of dose rate. This minimum thickness is calculated based on the dose rate at 2m away from the package since this dose rate relative to limit value is larger than one at the surface. In addition, these minimum thicknesses do not include the margin for error, so it should be considered at the stage of design in detail or manufacturing.

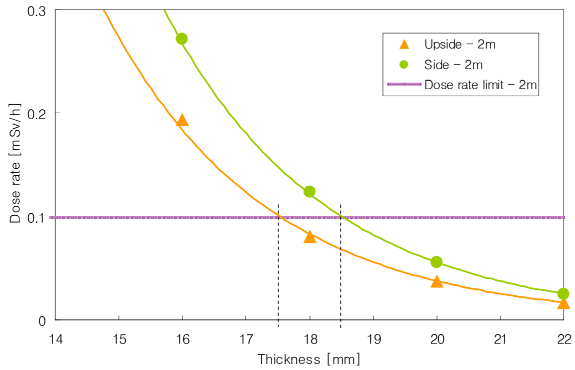

Figure 2. Dose rate at 2m away from the package versus thickness of shielding material of HIC transport package

Category	Min. thickness [mm]
Lid and bottom of package	17.5
Side of package	18.5

Table 4. Minimum thicknesses of HIC transport package

4. Conclusion

Radiation shielding analysis using MCNP code shows the minimum thicknesses of the shielding walls for the HIC transport package. However, many conservatisms and assumptions are considered in the calculation, the calculated minimum thickness could be much thicker than required one in the real. For the safety and economical efficiency, further calculation with less assumption and accurate radiation source term could help reduce the risk from the improper assumptions or the cost due to unnecessarily thicker shielding walls.

REFERENCES

[1] Ministry of Science and Technology, Regulations for Technical Standard for Radiation Safety Management and etc., 2001
[2] Ministry of Science and Technology, Notification from Ministry of Science and Technology No. 2001-23, Regulations for the Packaging and Transport for Radioactive materials, 2001
[3] International Atomic Energy Agency, IAEA Safety Standard Series, Regulations for the Safe Transport of Radioactive Material, Safety Requirements No. TS-R-1, 2005 Edition
[4] International Atomic Energy Agency, IAEA Safety Standard Series, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material, Safety Guide No. TS-G-11, Vienna, 2002
[5] Nuclear Engineering & Technology Institute, Report for Conceptual Design for Spent Resin Transport Cask, 2007
[6] X-5 Monte Carlo Team, MCNP-A General Monte Carlo N-Particle Transport Code, Version 5, 2003