Transactions of the Korean Nuclear Society Spring Meeting
Gyeongju, Korea, May 29-30, 2008

Preliminary Analysis of Radiation Shielding for HIC Transport Package Under the Hypothetical Accident Conditions

†Jaehoon Ko, Dohyung Kim, Unjang Lee, and Kyu-Sup Choi

Korea Nuclear and Engineering and Service Corporation
210-2 Hapdong Bldg., Yangjae-Dong, Seocho-Gu, Seoul 137-130, Korea
†Corresponding author: kojh4@kones21.com

1. Introduction

A radiation shielding analysis under the hypothetical accident condition 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 plat sites to disposal repository. Radiation source term is first carefully determined from the safety analysis reports related to HIC for appropriate calculation. And then MCNP5 is performed to obtain the minimum crevice between package lid and body, which meets the dose rate limit under the hypothetical accident conditions. Standards and codes of radiation shielding analysis related to the hypothetical accident condition are prescribed in Korea Nuclear Law, IAEA Safety Standards Series for Radioactive Material Transport and US 10CFR Part 71.

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) are in Table 1.

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

Category	Dose rate limit [mSv/hr]
Under normal transport conditions	
• Surface of the package	2
• 2m away from the package	0.1
Under the hypothetical accident condition	
• 1m away from the package	10

Under the Normal transport condition, dose rate at the surface of the HIC package is limited to 2mSv/hr and one at any point 2m away from the package is limited to 0.1mSv/hr. Under the hypothetical accident condition, dose rate at any point 1m away from the HIC package is limited to 10mSv/hr.

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 shielding calculations using Mote 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
- Detector positions

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.

For the hypothetical accident condition, the crevice between the lid and body is gradually increased to find the maximum gap which can meet the regulation limit.

3.2 Geometry and materials

The schematic geometry of HIC transport package for MCNP calculation is in Figure 1. It is in cylindrical shape and has a lid on the top. This cask can carry a HIC or spent resin in it. It is assumed that Impact limiters are completely destroyed and male screws of the cask lid are loosed. Therefore the MCNP5 models of the HIC transport package accident conditions have these zones replaced with air.

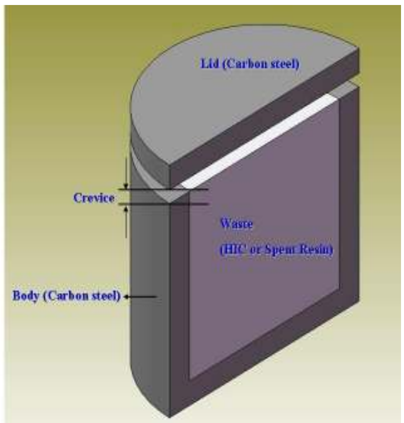

Figure 1. Schematic figure of HIC Transport package under hypothetical accident condition

All metals used in the calculation are assumed to be carbon steel as shielding materials for HIC transport packages. 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.

Table 2. Composition of materials for calculation

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) Ni(3.94) Mo(0.21) P(0.02) Cr(0.28) Nb(0.03) V(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)

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 rays, So the radioisotopes emitting gamma ray are considered as radiation sources in this calculation. And also gamma isotopes with major gamma isotopes are neglected. Table 3 shows the radioisotopes with gamma ray used in the calculation.

Table 3. Gamma ray source

Isotope	Energy [MeV]	Yield	Activity[Bq]
Co-58	0.0811	0.995	1.3048E+12
Co-60	1.173	0.999	4.2940E+12
	1.332	0.999	4.2940E+12
Cs-137	0.661	0.851	3.1796E+12
Ce-144	0.134	0.1109	3.5904E+09
Total			1.3076E+13

MCNP5 was used to calculate doses at the various desired locations. It calculates photon flux which can be converted into dose by the use of dose conversion factors. The dose conversion factors used in this calculation was taken from ICRP-74(1995)

3.4 Detector positions

In this MCNP calculation, virtual detectors (i.e. F5 tally card) are placed at the considering points to obtain the information fort dose rate. 15 detectors are placed at the points 1m away from the surface. These detectors are placed on the line which is parallel to the height direction of the package, and they are equally distributed from the crevice.

3.4 Results

The calculation results are shown in Fig. 2. It shows the crevices between package lid and body, which are satisfying regulatory limit. Under the hypothetical accident condition, allowable size of crevice between lid and body is about 6cm.

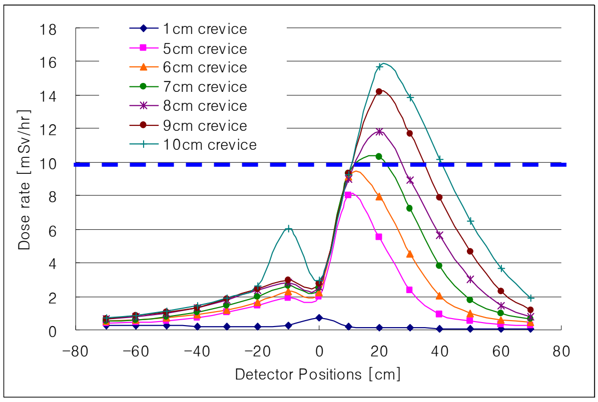

Figure 2. Dose rate at 1m away from the HIC Transport package under the accident condition

4. Conclusion

After the previous radiation shielding analysis for HIC transport package under normal transport condition, shielding analysis under accident condition is performed in the report. The allowable maximum crevice size of this package is about 6cm under the accident condition.

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
[3] International Atomic Energy Agency, IAEA Safety Standard Series, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material, Safety Requirements No. TS-R-1, 2005 Edition
[4] Nuclear Engineering & Technology Institute, Report for Conceptual Design for spent Resin Transport Cask, 2007
[5] X-5 Monte Carlo Team, MCNP-A General Monte Carlo N-Particle Transport Code, Version 5, 2003