COMPARISON OF THE MONTE CARLO CODES MCNP AND MONACO IN SHIELDING CALCULATIONS FOR TRANSPORT/STORAGE CASKS

H. Taniuchi
Transnuclear, Ltd.
1-18-16, Shinbashi, Minato-ku, Tokyo, 105-0004, Japan

ABSTRACT

The Monte Carlo code for shielding calculations of transport/storage casks, such as the MCNP and MONACO codes which are included in the SCALE code system, are compared in this paper. The comparison includes the calculation with both a simplified and a more detailed cask model to check the difference between the two codes concerning the capability of input data and the trend of the obtained results. The results show that both codes are very useful and obtain reasonable results within the reasonable CPU time. The MONACO code is better to get reasonable results with shorter CPU time than MCNP code when the input data is used in a simplified case. When the input data becomes more complicated the MCNP code is better. The obtained neutron/gamma dose rates are reasonably similar to each other.

INTRODUCTION

To design practical and efficient transport/storage casks for loading a higher neutron and gamma source intensity such as spent fuels, the shielding calculation is very important and Monte Carlo codes may be the most powerful tools for the calculation. So far, there are still difficulties for many users to obtain reasonable and reliable results without many trials. In the worst case, it is impossible to get reliable results. Thus, it is important to know the right method of Monte Carlo codes.

MCNP[1)] is the most popular Monte Carlo code and is used throughout the world for many applications. MCNP was developed by Los Alamos National Laboratory (LANL). MCNP uses continuous-energy nuclear and atomic data libraries. With respect to the shielding calculation for transport/storage casks, several reports exist. For example, the report on a comparison with measurements by the same author.[2)]

In 2009, the ORNL released a new variance reduction option called MAVRIC using MONACO. MONACO is not so often used. Based on MORSE code, it may be advantageous for shielding calculations because the MONACO code belongs to the SCALE 6[3)] code system which was developed as a licensing code system for casks originally funded by the NRC.

In this study, two major Monte Carlo codes, MCNP and MONACO, are compared based on the author's experiences. Therefore, the following discussion should be considered as a reference because some points may be not correct while others have already been improved by the code developers.

POINTS FOR APPLYING MONTE CARLO CODES FOR CASK SHIELDING CALCULATIONS

The important points for using Monte Carlo codes for shielding calculations are:

1) easy in making model configurations and easy in identifying geometric errors
2) ease to introduce effective variance reduction techniques
3) ease to judge the reliability of the results
4) reasonable total calculation time including input preparation.

For these points, MCNP and MONACO have the following characteristics summarised in Table1.

1) Model configuration including source and tally definition

MCNP has a long history of development, and the input data has been improving for a long time. Therefore, numerous options for geometry configuration and source definition with good reliability exist. On the other hand, MONACO is new code with few options even though, it is a very popular code with MORSE as base. The MONACO option is not developed enough for the cask shielding calculations yet. For example, following points should be modified for easier input for cask shielding calculations:

- Geometry: MCNP can define cells by using any number of surface cards or combinatorial-geometry-like macrobodies. MONACO can use similar geometry called General Geometry Package (SGGP), but the flexibility is limited compared to MCNP.

- Source: For simulating the burnup profile of neutron source distribution, MCNP can define as it is. However, the spatial distribution is not as well defined by MONACO because uniform distribution is only currently allowed. This point is very inconvenient for shielding calculations with used fuels. But, the next release of SCALE6.1 is planned to improve this aspect.

- Tally: To define volume tallies, MCNP can use any coordinate, but MONACO can only use Cartesian. The shape of used fuel casks is usually cylindrical, in which case the tally should be defined by cylindrical Polar Coordinates. Ring detectors can be prepared in MCNP but not in MONACO.

The advantage of MONACO is that definition of these data is easy to understand. MCNP can handle any feature, but complicates the input of MCNP and users may use it in an unintentional way.

2) Check of input error especially for geometric errors

Two calculation modules are under preparation to verify the geometrical data and detect eventual errors. For the MCNP code, Visual Editor will be used and for the MONACO code, KENO3D. Both modules will be very useful.

3) Variance reduction technique

A general variance reduction technique can be used for both codes, MCNP and MONACO. The most significant and important difference is that MCNP uses WWG(Weigh Window Generator)whereas MONACO uses MAVRIC(Monaco with Automated Variance Reduction using Importance Calculations). WWG does not generate appropriate Weight Window(WW)

automatically before the calculation, but is expected to generate better WW during the calculation. This is a good option, but it depends on the original calculation input. If the first assumption for variance reduction is not correct, the expected WW cannot be easily obtained.

In contrast, MAVRIC automatically prepares energy dependent space mesh importance. More precisely, MAVRIC is based on the CADIS (Consistent Adjoint Driven Importance Sampling) methodology, which uses an importance map and biased source that work together. MAVRIC will automatically perform a coarse mesh, three-dimensional, discrete ordinate calculation using Denovo to determine the adjoint flux as a function of position and energy. This adjoint flux information is then used by MAVRIC to prepare a space and energy-dependent importance map (i.e. very detailed weight windows). This is the most advantage of MONACO point for shielding calculations. Moreover, MAVRIC can prepare an importance map for multi-detector which is very convenient.

4) Reliability of the results

To check the reliability of the calculated results, MCNP prepares ten statistical check items. MONACO does not currently prepare as much , but is planed to have similar check items in the next release of SCALE6.1.

5) Total calculation time including input preparation

To compare the calculation time of MCNP and MONACO, sample problems have been prepared with calculations.

Table 1. Comparison of input data option for MCNP and MONACO

Item		MCNP	MONACO
Library		Continuous	Multi-group
Geometry	Definition	Easy/ complicated	Easy
	Expression	None	Limited
Source	No. of source region	None	Limited
	Source distribution	None	Uniform only (under creation)
	Definition	Flexible, but complicated	Limited, but easy
Tally	Coordinate	None	Cartesian only
	Conversion factor	Not prepared	Prepared
	Definition	Flexible, but complicated	Limited, but easy
Variance Reduction		WWG (Weight window generator)	CADIS
		For only one detector	For multi-detectors
Estimation of Error		Ten statistical checks	Plan to prepare statistical checks
Visualization of outputs		Particle display Tally plots(Graph)	Mesh tally viewer Importance map

COMPARISON OF CALCULATED DOSE RATES

Two calculation models are prepared to check the usabilioty of both codes concerning the points mentioned above. The first one is the simple cask geometry that assumes the neutron shield region including copper fins is homogeneous. The second one is the detailed cask geometry with explicitly defined copper fins in the neutron shield region.

In both calculations, the following conditions are considered:

1) Source: the neutron source spectrum is assumed as Pu-239 and is specified by Watt spectra equation. The total neutron source intensity is set to 1.0E9. the FP gamma source spectrum is assumed to represent a BWR used fuel cooled for 18 years specified by a histogram as shown in Table 2. The total FP gamma source intensity is set to 1.0E16. The source region is assumed as a ring shape in the cask cavity to reduce the calculation time.

2) Detector: Point detector (1m from the cask surface at axial center) and large surface ring detector (cylindrical surface detector with 2m axial height at 1m from the cask surface at axial center) are selected. The surface detector is not prepared in MONACO, so the surface detector–like volume detector must be created this time.

Table 2. Gamma source spectra for shielding calculation

Energy Group	Upper Energy (eV)	Source Intensity
6	4.00E+06	7.59E-07
7	3.00E+06	5.88E-06
8	2.50E+06	1.24E-04
9	2.00E+06	6.45E-04
10	1.66E+06	1.02E-02
11	1.33E+06	2.44E-02
12	1.00E+06	7.96E-02
13	8.00E+05	3.13E-01
14	6.00E+05	5.56E-01
15	4.00E+05	1.63E-02
Total	-	1.0

Simplified uniform model

The first model is a very simple cylindrical cask assuming a used-fuel source. As shown in Fig. 1, the model is an axially uniform cylindrical geometry.

a) MCNP calculation

There are two choices for applying WWG to cask shielding calculations. One is cell-based WWG and the other is superimposed importance mesh-based WWG. To save calculation time, and avoid input geometry error, the superimposed-importance mesh-based WWG is preferable. MCNP prepares both Cartesian and Cylindrical superimposed mesh geometries, the cylindrical mesh is easier and better to apply to cylindrical cask geometry.

Neutrons: To apply WWG, the original cell importance needs to be defined first. To check the effect from the original cell importance, the neutron dose rate calculations are performed with three different cell importance sets. The results summarized in Table 3 show that there is no significant difference in the obtained dose rates, and reasonable neutron dose rates are obtained even though only one calculation is done without using WWG in any case. It is not recommended to use WWG when the IMP is set to 1.0 for all cell-importance points of origin because the CPU time is very

long. The neutron calculation is not sensitive enough for an initial estimation of cell-importance when moderate cell-importance is prepared.

Gamma: The method applied to neutron calculations is not applicable for gamma calculations. The gamma attenuation in a cask is very rapid and when a superimposed importance-mesh is used, the number of cells is not sufficient to define appropriate cell importance for gamma attenuation. One solution is to input the WWINP by hand which is not so easy. Therefore, another easier solution is to make a few runs to get good WW in the WWINP. In the case of the calculated dose rates shown in Table 3, two runs of 100 minutes and 500 minutes respectively were first performed to get good WW in WWINP. The total CPU time for gamma dose rate calculation is thus approximately one order longer than that for neutron dose rate calculation.

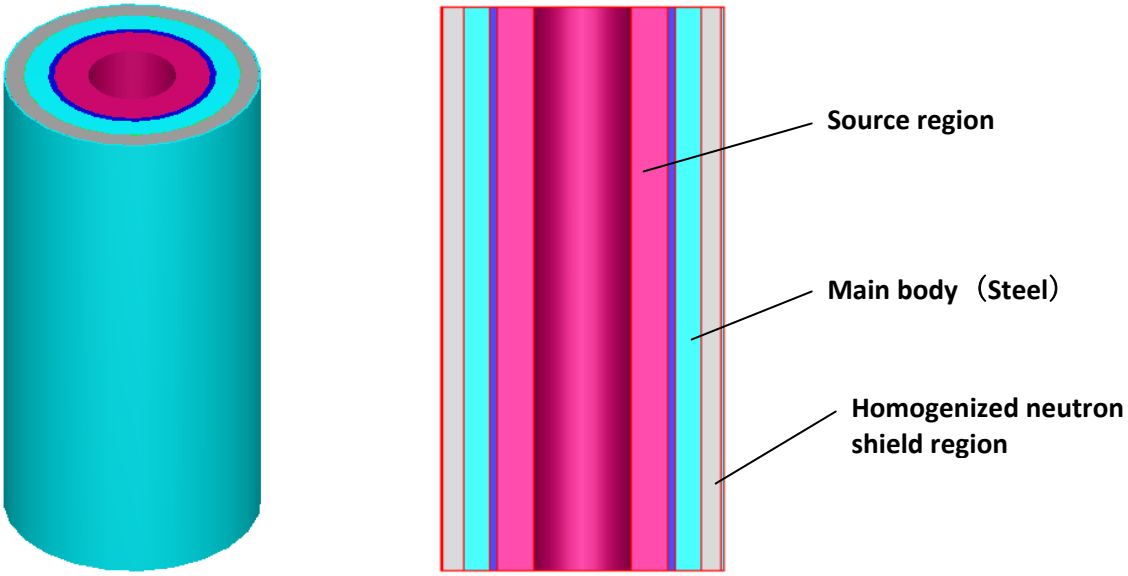

Figure 1. Simplified uniform model (height: 5m, diameter: 2.6m)
Picture from KENO3D

Table 3. Effect of setting cell importance in MCNP calculation with simplified model

Source type	Variation of cell importance for first run	CPU time (min.)	history	Point detector		Surface Ring detector	
				Dose rate	Error*)	Dose rate	Error
Neutron	Imp=1(all cell)	477	50000000	0.63	0.033(10/10)	0.62	0.022(10/10)
	Imp=1 to 5000	100	105428	0.58	0.038(10/10)	0.63	0.017(10/10)
	Imp=1 to 128	100	2774368	0.63	0.030(10/10)	0.63	0.017(10/10)
Gamma	Imp=1 to 5000	500	368253756	6.65	0.051(8/10)	6.76	0.023(9/10)
	(longer time)	-	4579748533	7.16	0.018(10/10)	6.93	0.008(10/10)

*) The number in brackets shows the results of 10 previous statistical checks.

b) MONACO calculation

When MAVRIC is used for input of MONACO code, several options for variance reduction techniques are prepared. The CADIS method is most often used, and it is this method which was chosen for this study.

Firstly, the effect of the number of energy groups for the cross-section library for MONACO was checked because MONACO uses a multi-group cross section library. Two libraries, 200n+47g and 27n+19g are included in SCALE6. Table 4 shows that the calculated dose rate using the 27n+19g library is the almost same as that of the 200n+47g library for gamma dose rates, but 60% higher for neutron dose rate in this case. This difference may be caused by the structure of cross-section library. The neutron group structure of the 27n+19g library in higher energy is too wide to represent as the effective cross-section data. For example, there are only 5 neutron groups for energy higher than 1.0MeV. When the number of the energy group is small enough such as the 200n+ 47g library, this problem does not occur, but the 27n+19g energy group structure is not small enough for a fission spectra.

The effect of SN calculation parameter for Denovo is then checked. All Denovo calculations are performed with a P_1S_4 set, except the reference calculations which are done with a P_3S_8. Because of the limitation of core memory, the smaller number of mesh is used for P_3S_8 calculation. The results show that there is not so much difference in the dose rates, but the time for neutron calculation with Denovo is significantly longer for the P_3S_8 calculation. Therefore, P_1S_4 for Denovo calculation is adequate enough.

After that, the effect of mesh size is compared. The total number of mesh is approximately 100,000 for the reference case, around 70,000 for the uniform mesh case and 180,000 for the large number of mesh case. The results show that the difference is negligible. The uniform mesh within the cask geometry unexpectedly corresponds well with other cases. This is preferable because the input becomes easier in this case.

Table 4. Effect of setting mesh in MONACO calculation with simplified model

Source type	Library and number of mesh		CPU time*) (min.)	history	Point detector		Surface Ring detector	
					Dose rate	Uncertainty	Dose rate	Uncertainty
Neutron	Reference case	27n+19g	49(+11)	6000x100	0.97	0.015	0.97	0.006
		200n+47g	42(+72)		0.59	0.020	0.59	0.009
		P3S8(200n)	44(+145)		0.58	0.021	0.59	0.009
	Uniform mesh		46(+53)		0.63	0.020	0.60	0.010
	Large number of mesh		39(+110)		0.61	0.019	0.60	0.008
Gamma	Reference case	27n+19g	169(+5)	250000x100	6.78	0.011	6.70	0.004
		200n+47g	229(+21)		6.73	0.011	6.67	0.004
		P3S8(47g)	205(+28)		6.53	0.011	6.69	0.005
	Uniform mesh		235(+19)		6.70	0.011	6.65	0.004
	Large number of mesh		233(+31)		6.76	0.010	6.73	0.004

*) The number in brackets shows the additional CPU time such as Denovo calculation

c) Comparison of results

Table 5 shows the comparison of the obtained dose rates by MCNP and MONACO for the typical cases from table 3 and 4. With respect to neutron sources, the obtained dose rates and CPU time are very similar and the level of error or uncertainty is the same. On the other hand, there is a large difference between MONACO and MCNP for gamma calculations. The obtained dose rates are very similar, but MCNP needs twice the CPU time as that of MONACO for the calculation, and MCNP needs more time to prepare moderate WW in WWINP. About five times CPU time is needed for MCNP in this case. Also, MONACO has less uncertainty with shorter CPU time. This is because it is difficult to prepare the most suitable WW for MCNP calculations without any previous calculations. Of cause, a good tool to prepare estimated WW will obtain reasonable dose rates by MCNP quickly.

Table 5. Comparison of MCNP and MONACO with simplified model

Source type	Item	MCNP		MONACO	
		Dose Rate	Error	Dose Rate	Uncertainty
Neutron	Point Detector	0.63	0.030	0.59	0.020
	Surface Ring Detector	0.63	0.017	0.59	0.011
	CPU Time	100		42(+72)	
	No. of Calculation	1		1	
Gamma	Point Detector	6.65	0.051	6.73	0.011
	Surface Ring Detector	6.76	0.023	6.67	0.004
	CPU Time	500(1100)		229(+21)	
	No. of Calculation	2+1		1	

Detailed fin model in neutron shield region

Figure 2 shows the second calculation model. The overall dimensions and materials are the same with the simplified model shown in Figure 1, but the fins in the neutron shielding region are explicitly modelled here. The source conditions and tally setting are the same as for the simplified model.

The comparison of the calculated dose rates is shown in Table 6. With respect to the obtained dose rates, the values are almost same for both cases, but the CPU time is different. The MCNP CPU time is the same in the simplified model except that an additional run is necessary to prepare a proper WW in WWINP. The preparation time is then five times longer for the case of MONACO. With respect to gamma sources, the trend is the same as the neutron sources. The MCNP CPU time is three times longer than the simplified model, but ten times longer for MONACO. The cause of this trend has not been investigated so far, but it can be said that the CPU time for MCNP is minimally affected by the complexity of geometry input but that MONACO in contrast is significantly affected.

CONCLUSIONS

In this study, two principle Monte Carlo codes, MCNP and MONACO used for cask shielding calculations are compared based on the author's experience. It is confirmed that both codes are good enough to calculate dose rates, except for the neutron calculation by MONACO with a

27n+19g cross-section library. This library may have too wide an energy boundary to treat fission spectra for the neutron source.

With respect to neutron shielding calculations of cask models, the reliability and the total calculation time of both codes are reasonable enough to apply to cask shielding calculations. On the other hand, a considerable longer calculation time is necessary to obtain a reliable gamma dose rate because the attenuation of the gamma dose rate inside the cask is several levels higher than that of the neutrons dose rates. Regarding gamma shielding calculations, the variance reduction method in MAVRIC seems more powerful than that in the Weight Window Generator in MCNP when the model is simple. But, for detailed models, the CPU time is the same to get reliable gamma dose rates.

For the user of the Monte Carlo codes, the best mixture of MCNP and MONACO is preferable.

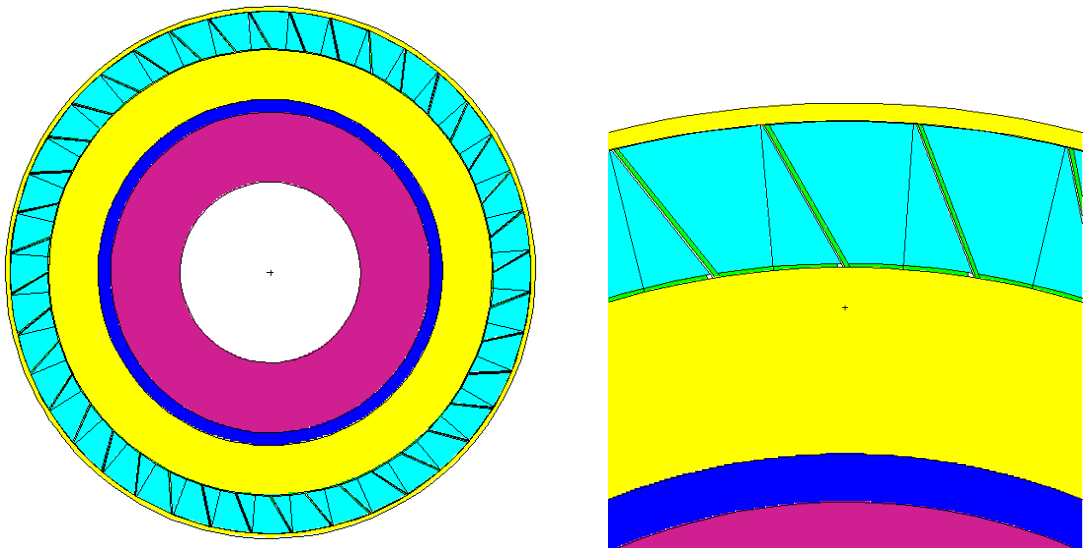

Figure 2. Detailed fin model in neutron shield region (height: 5m, diameter: 2.6m) Picture from MCNP Visual Editor

Table 6. Comparison of MCNP and MONACO with detailed model

Source Type	Item	MCNP		MONACO	
		Dose Rate	Error	Dose Rate	Uncertainty
Neutron	Point Detector	0.66	0.014(8/10)	0.64	0.044
	Surface Ring Detector	0.66	0.008(8/10)	0.62	0.019
	CPU Time	100(+100)		67+600	
	History	19133462		6000x26	
	No. of Calculations	2		1	
Gamma	Point Detector	6.95	0.042(9/10)	6.52	0.032
	Surface Ring Detector	7.01	0.015(10/10)	6.82	0.012
	CPU Time	1500(+600)		15+2030	
	History	641118619		100000x31	
	No. of Calculations	2+1		1	

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