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STUDY OF ANALYSIS METHODS OF SHIELDING CALCULATION CODES FOR CASKS

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Abstract

It is common in Japan to apply QAD-CGGP2R, DOT3.5 and DORT to safety analysis for transport and storage cask, and MCNP code is being introduced recently.

In order to study the features of each analysis code, using a cask shielding benchmark problem, investigations are carried out in this paper concerning the following subjects.

a) Influence of difference of nuclear data library on calculation results (DORT).
b) Influence of number of angular quadratures on ray effect (DORT).
c) Influence of homogenization of fuel region on calculation results (QAD-CGGP2R, MCNP)
d) Influence of difference of analysis code on calculation results (QAD-CGGP2R, DORT, MCNP)
e) Effectiveness of the latest method for variance reduction parameter setting (ADVANTG + MCNP)

The results of the above investigations show features of each analysis code and that attentions shall be paid to setting of parameters of the analysis code when utilized. In addition, the latest knowledge concerning effectiveness of ADVANTG recently released from by ORNL (Oak Ridge National Laboratory) for variance reduction parameter setting of MCNP has been obtained.

1. Introduction

It is common in Japan to apply QAD-CGGP2R [1], DOT3.5 [2] and DORT [3] to safety analysis for transport and storage cask, and MCNP [4] code is being introduced recently.

It is necessary to understand that each analysis code has specific features and should be utilized appropriately according to the purpose of the analysis. In this paper, various investigations are carried out in order to contribute to the improvement of analysis methods applied to transport and storage cask.

2. Details of the study

In order to achieve the purpose mentioned above, parameter survey calculations are carried out and results are shown below.

2.1 Analysis model and conditions

Various calculations have been carried out by using problem1 and problem4 selected among the NEA-CRP (Nuclear Energy Agency, Committee on Reactor Physics) shielding benchmark problems [5].

The analysis model and conditions are shown in the followings.

(1) Analysis model

The analysis model consists of body, basket and fuel assemblies.

The body is cylinder shape with a 380mm thick shell (see Figure 2.1.1). The basket has a lattice shape with plates of 10mm thickness (see Figure 2.1.2). The fuel assembly has 15 x 15 arrays with 3420mm of active length (see Figure 2.1.3 and 2.1.4).

(2) Analysis conditions

- 5 fuel assemblies are loaded.
- Source intensity: Neutron 1×10^{9} n/sec, Gamma 5.0×10^{16} photons /sec

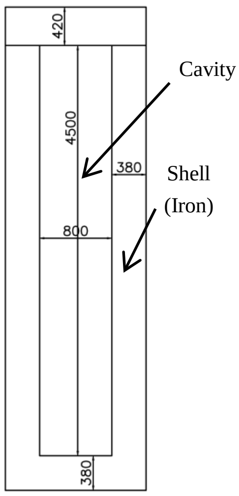


Figure 2.1.1 Body

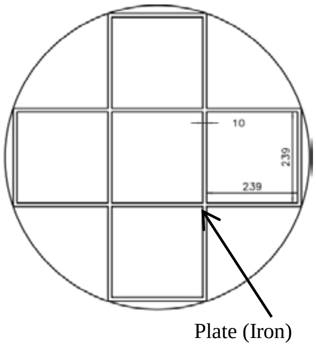


Figure 2.1.2 Fuel Basket

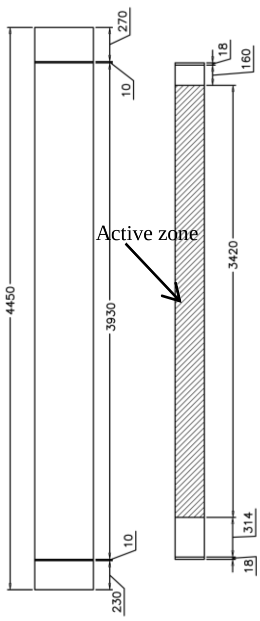


Figure 2.1.3 Fuel Assembly

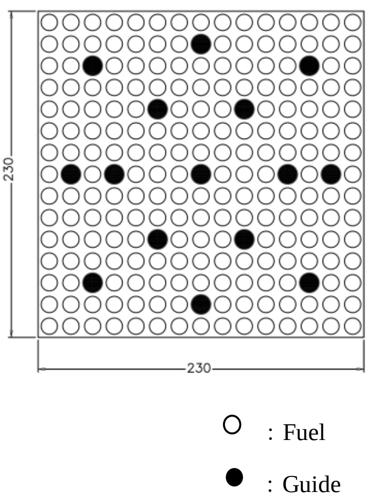


Figure 2.1.4 Arrangement of fuel rods

2.2 Analysis results and Discussion

Analysis results obtained by using the model mentioned above and their comparison and discussion are shown in the followings.

2.2.1 Influence of difference of nuclear data libraries on calculation results (DORT)

Different nuclear libraries are available for shielding analysis by DOT3.5, DORT, and MCNP. It is important to choose appropriate nuclear data, because different energy group structure and/or nuclear data version might give different results. In this paper, calculations to study the influence of different nuclear libraries are carried out by using DORT code with four kinds of nuclear data.

The source region is homogenized and dose rate of neutron at 1m from shell surface are detected. The results are shown in Table 2.2.1.

Table 2.2.1 Calculation result (at 1m from side surface)

Cross section Library	Energy Groups	Dose rate [μ Sv/h]	Ratio to SCALE ENDF/B-VII
DLC-23F[6] ENDF/B-II	22n-18g	284	0.90
SCALE5[7] ENDF/B-IV	27n-18g	331	1.04
SCALE5[7] ENDF/B-VI	200n-47g	328	1.03
SCALE6[8] ENDF/B-VII	200n-47g	317	1.00

(1) Comparison of results

- The results of SCALE libraries are about 12–to 17 % greater than the result of DLC-23F library (first version of the ENDF/B library).
- The result of ENDF/B-IV which has a small number of energy groups are almost same as the results of ENDF/B-VI and ENDF/B-VII which possess more energy groups.
- The result of ENDF/B-VII (latest version of the ENDF/B library) is almost same as the result of ENDF/B-VI.

(2) Discussion

- In the 0.5MeV to 0.9MeV neutron energy region, the neutron dose rates of SCALE are greater than the neutron dose rate of DLC-23F as shown in Figure2.2.1. The reason is that a self-shielding factor is absence in the DLC-23F library [9]. However, it is no problem to apply the DCL-23F library to safety analysis for transport and storage cask, because the dose rate obtained with this library show good agreement with the results of MCNP calculations [10] [11].
- SCALE has variation such as ENDF/B-IV and ENDF/B-VI or ENDF/B-VII, and they give small different results.
- The macro cross sections of the flask body (iron) calculated with ENDF/B-VI are completely same with those calculated with ENDF/B-VII. Therefore, small different results may be caused by the differences of macro cross section of the homogenized fuel region.

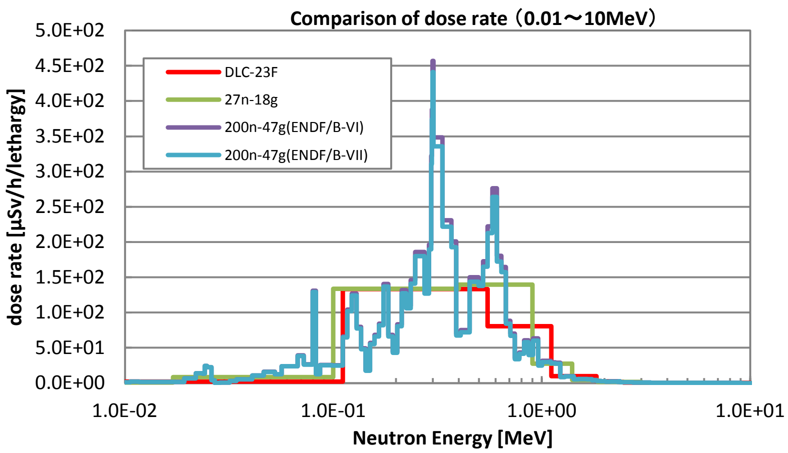


Figure 2.2.1 Comparison of dose rate

2.2.2 Influence of number of angular quadratures on ray effect (DORT)

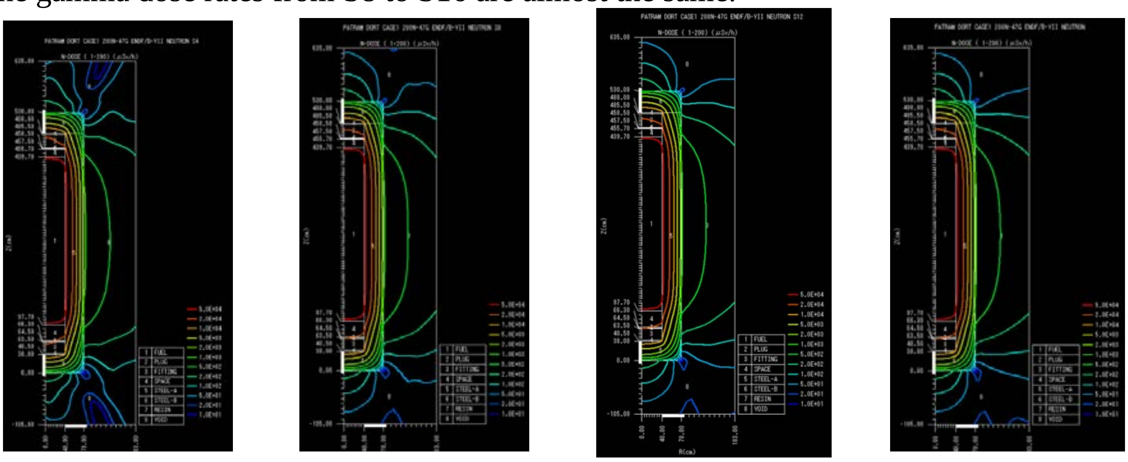
The discrete ordinate code DORT has the advantage for saving calculation time comparing with the Monte Carlo code MCNP. However, DORT has the disadvantage that ray effect could be given when the discretization accuracy is not sufficient. In this paper, calculations to study the influence on ray effect of numbers of angular quadratures are carried out by using 4 cases of angular quadratures. The source region is homogenized and dose rate of neutron at 1m from bottom surface are detected. ENDF/B-VII libraries are used for this calculation and the expansion degree P_l is 3. The results are shown in Table 2.2.2 and Figure 2.2.2.

Table 2.2.2 Calculation result (at 1m from bottom surface)

Dose Rate [μ Sv/h]	Number of angular quadratures			
	4	8	12	16
Neutron	38.4	38.5	36.9	36.4
Gamma	1.0	2.5	2.4	2.6

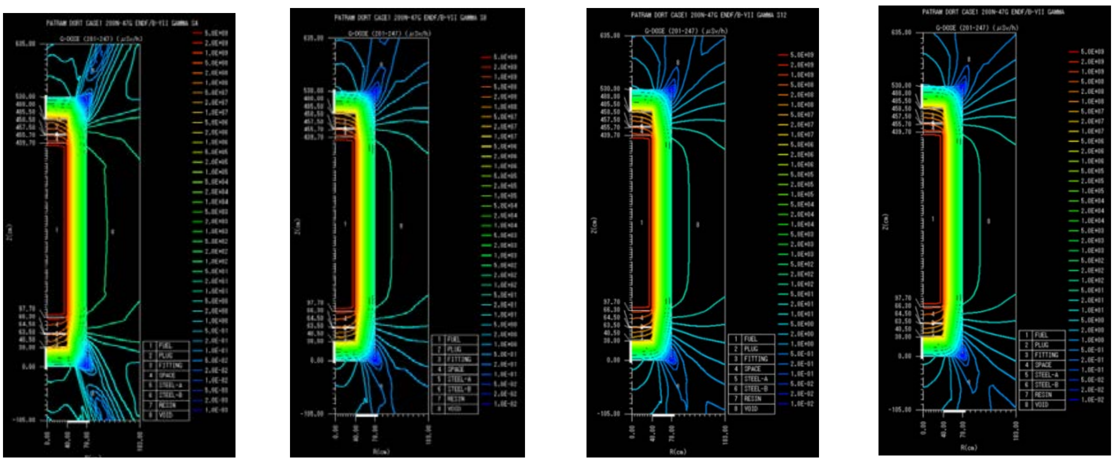
(1) Comparison of results

- Ray effect is decreasing according to increase of the number of angular quadratures as shown in Figure 2.2.2.
- The neutron dose rate is slightly decreasing according to increase of the number of angular quadratures.
- The gamma dose rates from S8 to S16 are almost the same.



S4 S8 S12 S16

Figure 2.2.2(1) Neutron dose equivalent rate distribution



S4 S8 S12 S16

Figure 2.2.2(2) Gamma dose equivalent rate distribution

(2) Discussion

- Both the gamma ray effects and the neutron ray effect decrease according to increase of the number of angular quadratures. The reason is as the angle difference between flights paths directions decrease, the model approaches a more accurate continuous model.
- The neutron dose rate of S8 is slightly greater than the one of S16. The gamma dose rate of S8 is in good agreement with the one of S16. Therefore, the number of angular quadratures of S8 can be applied to safety analysis for transport and storage cask.

2.2.3 Influence of homogenization of fuel region on calculation results (QAD-CGGP2R, MCNP)

It is common to simplify the calculation model by homogenization of fuel assembly region in order to shorten calculation time and to obtain conservative results. In this paper, calculations to study the influence of modeling of source region are carried out with simplified model and actual detail model on QAD-CGGP2R and MCNP.

2 types of homogenization are studied. The first one is named model-1, every parts inside the cavity (fuel and basket) are homogenized. The second one named model-2, some parts inside basket (fuel assembly) is homogenized. A detail model has actual structure of basket and fuel assemblies. These models are shown in Figure2.2.3.

The MCNP code version is 6.1 and the ENDF/B-VII library is used. Atomic densities of each model are shown in Table2.2.3. The results are shown in Table2.2.4.

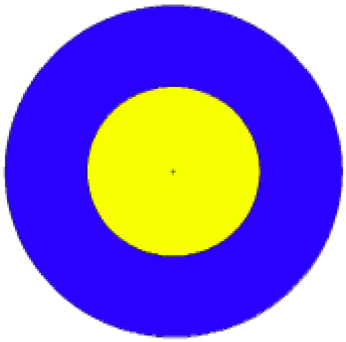


Figure 2.2.3 (1)
Homogeneous model-1

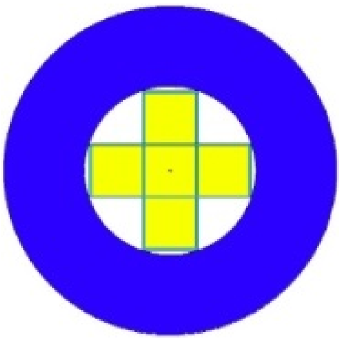


Figure 2.2.3 (2)
Homogeneous model-2

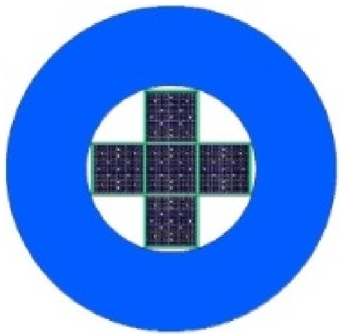


Figure 2.2.3 (3)
Detailed model

Table 2.2.3 Atomic density (at 1m from side surface)

Homogeneous model-1 (Inside the cavity)	**Homogeneous model-2 (Inside compartment)**	**Detail model (UO2 pellet)**
2.21 g/cm^3	3.08 g/cm^3	9.67 g/cm^3

Table 2.2.4 Calculation result

Dose rate [μ Sv/h]	**①Homogeneous model-1**	**②Homogeneous model-2**	**③Detail model**	**Ratio**	
				②/①	**③/①**
Neutron MCNP	387	373	376	0.96	0.97
Gamma MCNP	31	27	26	0.87	0.84
Gamma QAD-CGGP2R	42	39	—	0.93	—

(1) Comparison of results

- The neutron dose rate obtained by MCNP with the homogeneous model-1 is the highest, and about 4% greater than the homogeneous model-2.
- The gamma dose rate obtained by MCNP with the homogeneous model-1 is the highest and about 13% greater than the homogeneous model-2. The gamma dose rate obtained with QAD-CGGP2R is also about 7% greater than the homogeneous model-2. Conservativeness given by homogenization is higher on gamma-ray than on neutron.
- Both neutron and gamma dose obtained by MCNP with the homogeneous model-2 are almost the same with the detailed-model.

(2) Discussion

- The reason why neutron dose rate obtained by MCNP with the homogeneous model-2 is lower than the dose rate with model-1 is a effect of the basket plates being modeled outside of fuel region.
- The reason why gamma dose rate obtained by MCNP and QAD-CGGP2R with the homogeneous model-2 is lower than the dose rate with model-1 is the same as neutron dose rate mentioned above. The results of MCNP show that conservativeness of homogenization is more significant in case of gamma-ray than in case of neutron. The reason is that the basket plates (stainless steel) modeled separately is more effective for shielding against gamma ray than against neutron.
- The results obtained by MCNP show that the homogeneous model-2 and the detailed-model are almost equal. This means that homogenization inside basket could give enough accuracy.
- Both the neutron and gamma dose rate are the highest in case of the homogeneous model-1 on MCNP and QAD-CGGP2R. This means that homogenization of fuel and basket, which method is common for transport and storage cask, has enough conservativeness.

2.2.4 Influence of difference of analysis code on calculation results (QAD-CGGP2R, DORT, MCNP)

DOT3.5 code is generally applied to safety analysis of transport and storage cask in Japan, and it getting popular recently to use MCNP code for verification of the results. In this paper, calculations to study the influence of difference of analysis code on calculation results are carried out by QAD-CGGP2R, DORT, and MCNP.

The homogeneous model-1 is used for all calculations. ENDF/B-VII library is used for DORT and MCNP. The results are shown in Table 2.2.5.

Table 2.2.5 Calculation result

Dose rate [μSv/h]	QAD-CGGP2R	DORT	MCNP
Neutron	—	317	387
Gamma	42	29	31

(1) Comparison of results

- The gamma dose rate obtained by QAD-CGGP2R is greater than the gamma dose rate calculated by either DORT or MCNP.
- The neutron dose rate obtained by MCNP is about 20% greater than the neutron dose rate calculated by MCNP.

(2) Discussion

- The gamma dose rate obtained with QADCGGP2R is the highest. This result is caused by the build-up factor of QAD-CGGP2R, assuming an infinite medium.
- The neutron dose rate inside iron is studied as shown in Figure 2.2.4. The results of DORT are larger than the results of MCNP below an iron thickness of approximately 20cm, then becomes smaller. The similar phenomenon is presented in a different paper [12]. The cause is under investigation.

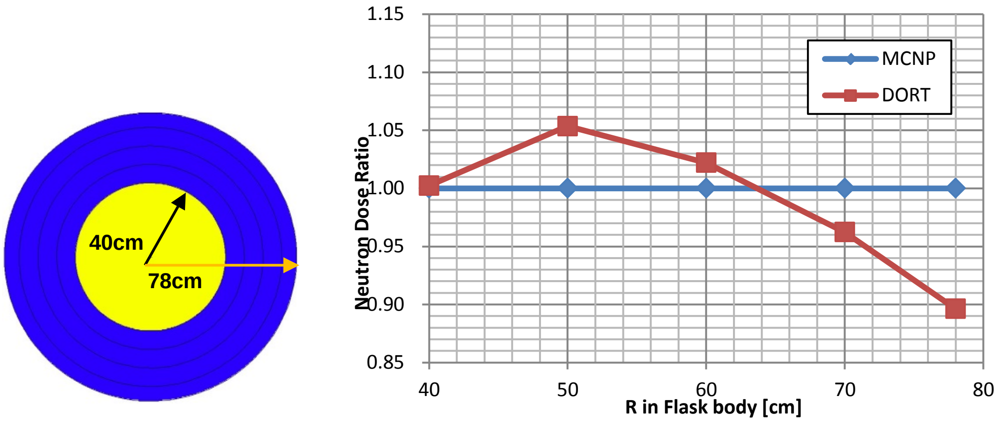


Figure 2.2.4 Comparison between DORT and MCNP

2.2.5 Effectiveness of setting the latest variance reduction parameter (ADVANTG + MCNP)

Effectiveness of the ADVANTG [13], which produces automatically reasonable variance reduction parameters and source bias, is studied by comparison with conventional methods requiring setting variance parameters manually.

CADIS method for a single tally and FWCADIS method for plural tallies are included in ADVANTG calculation.

The homogeneous model-1 is used for this calculation. The dose rate of neutron at 1m from side, top and bottom surface are detected. The MCNP code is version 5.1.6, which version can handle ADVANTG, and ENDF/B-VII library is used for nuclear data.

i) Conventional method :

In order to avoid deviation of calculation results by operator, the following steps of calculation are applied.

Each cell have “cell importance=1.0”, for variance reduction. By using the function of Weight Window Generator equipped with MCNP5.1.6, the calculation is continued until the statistical error is lower than 10%. By using Weight Window parameters obtained with the previous calculation, the time of the next calculation until the statistics error is lower than 5% is measured.

ii) CADIS method :

In case of CADIS method, only adjoint calculation is carried out at one point of target tally.

By using Weight Window parameters and source bias produced automatically by ADVANTG, the calculation is continued until the statistics error is lower than 5%.

iii) FW-CADIS method :

In case of FW-CADIS method, forward calculation is carried out firstly and adjoint calculation is carried out secondary because of generation of weight window value making plural tally as the target.

By using Weight Window parameters and source bias produced automatically by ADVANTG,

the calculation is carried out until the statistics error is lower than 5%.
The comparison of total calculation times and results are shown in Table 2.2.6 and Table 2.2.7.

Table 2.2.6 Comparison of total calculation time between conventional method and ANDVANTG

[min]

Calculation Method	Directions			TOTAL
	Top	Side	Bottom	
(i) Conventional Importance + wwg ⇒ WW	1038	401	2853	4292
(ii) ADVANTG CADIS ⇒ WW and source bias	82	73	103	258
(iii) ADVANTG FWCADIS ⇒WW and source bias	350	262	389	389*
Ratio (i)/(ii)	12.7	5.5	27.7	16.7
Ratio (i)/(iii)	3.0	1.5	7.3	11.0

* : Total calculation time is the calculation time of the neutron dose rate of bottom that requires the longest calculation time.

Table 2.2.7 Comparison of impact on neutron dose rates between calculation methods

[dose rate: μ Sv/h]

Calculation Method	Directions		
	Top	Side	Bottom
(i) Conventional Importance + wwg ⇒ WW	59.7 (1.69%)	397.9 (1.95%)	43.7 (3.23%)
(ii) ADVANTG CADIS ⇒ WW and source bias	56.2 (2.17%)	384.7 (1.43%)	45.8 (1.02%)
(iii) ADVANTG FW-CADIS ⇒WW and source bias	57.7 (1.76%)	381.9 (0.79%)	47.3 (1.88%)
Ratio (ii)/(i)	0.94	0.97	1.05
Ratio (iii)/(i)	0.97	0.96	1.08

( ) : statistical error

(1) Comparison of results

- CADIS method improves calculation times, which are about 1/6 in side direction and about 1/28 in bottom direction comparing with the conventional method.
- FW-CADIS method improves calculation times, which is about 1/11 in total times comparing with the conventional method.
- The difference of calculation result by calculation methods is slight and almost within a statistics error.

(2) Discussion

- ADVANTG provides high efficiency of calculation thanks to easy operation comparing with conventional method. Especially, FW-CADIS method is the most effective, because it can improve efficiency of calculation for any direction with only one calculation.
- The difference of calculation result by calculation methods is slight and almost within a statistics error. ADVANTG is very effective because it provides an objective setting of various reduction parameters, which has been process subjectively. This means that same result of the calculation can be obtained independent of the operator.

- An expected function in future is that priority can be assigned at evaluation of plural tallies. This could give more efficiency on calculation by using MCNP.

3. Conclusion

We can summarize the investigations presented in this paper as follow:

- For recent libraries, dose rates are independent of library version or energy group structure.
- The number of angular quadrature points of S8 can be applied to safety analysis for transport and storage cask.
- Homogenization of fuel and basket, a common method for transport and storage cask, is conservative enough.
- The results of DORT are larger than the results of MCNP below an iron thickness of approximately 20cm, then become smaller. The cause is under investigation.
- Effectiveness of ADVANTG is confirmed.

The results will be taken into consideration for our future safety analysis of casks.

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