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RATIONAL SHIELDING ABILITY EVALUATION FOR A MODULAR SHIELDING HOUSE

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ABSTRACT

In Japan, it is in the planning stage of building interim storage facilities for transportable storage casks. Nuclear spent fuels stored for a prolonged period of time in casks in such facilities would be transported to fuel reprocessing facilities without taking the lid off casks. The criterion of effective dose rate around the boundary of the interim storage facility would be 50μSv/year in Japan, which is one-fifth of the criterion applied in the United States. Though the capacity of the facility heavily depends on the dose rate, any rational methodologies to evaluate dose rate around the interim storage facility have not been developed, because the scale of the system is so large and numerous distributed radiation source terms make calculational procedure complicated. In this paper, we discuss rational methodology to evaluate dose rate at the facility boundary by using two approaches as follows; 1) a method taking into account "shade effect" which represents the self-shielding effect of interim storage casks, 2) utilization of newly-developed simplified code, MCNP-ANISN_W.

Considering “shade effect", some radioactive sources in the storage casks are surrounded by other casks in the interim storage facility, thus, the increasing number of the storage casks shield against the radioactive sources much more. Hence, it becomes possible to increase the number of allowable storage casks in the facility. Using the method of shade effect, the dose rate was calculated for a facility stored in 8 or 16 casks and the results of calculation were consistent with that of full Monte-Carlo calculation.

In this study, a simple neutron transport code, MCNP-ANISN_W, which incorporates one-dimensional transport code, ANISN, into Monte-Carlo code, MCNP, was developed. The contribution of the direct radiation was calculated by MCNP-ANISN_W. The effect of skyshine was separately calculated by using SHINE-III. The total dose of these results was consistent with that of the full Monte-Carlo calculation.

Results show that the simple evaluation technique and the code developed in this study would be useful for evaluation of dose rate around the boundary of the interim storage facility. Furthermore, the technique and code is applied to evaluate ships loading many casks.

INTRODUCTION

In Japan, spent fuels have been stored at spent fuel pools in the nuclear power plants, but the storage capacity of the pools comes to the critical limit. For this reason, Recyclable-Fuel Storage Company is planning construction of the interim storage facilities for the nuclear spent fuels. For the construction, it is necessary to design the shielding house to meet the criterion of effective dose rate around the boundary of the interim storage facility. While the facility capacity heavily depends on the dose rate, any rational methodologies to evaluate dose rate of the interim storage facility have not been established, because the scale of such a system is so large and numerous distributed radiation source terms make calculational procedure complicated.

In this research, using the "shade effect", the mutual shielding effectiveness among casks, the dose rate around the boundary of the facility was calculated for a facility which has 8 casks stored in 2 shielding houses or 16 casks stored in 4 shielding houses. The results of calculation were consistent with that of full Monte-Carlo calculation.

SHIELDING SYSTEM

At first, Monte Carlo Coupling Technique was used to analyze the shielding ability of the shielding house. The concept of this technique is shown in Fig.1. The modular shielding house in which two spent fuel transportable storage casks for one unit are installed is modeled, as shown in Fig.2. The main shielding structure of the interim shielding house consists of the steel-water-steel multilayer shielding system, so the shielding system is particularly effective for neutrons and gamma rays from the fission products. However, a part of the neutrons that penetrate the casks and also secondary gamma-rays produced $^{1}H(n,\gamma)$, $^{56}Fe(n,\gamma)$ reactions with thermal neutrons in the multilayer system stream into the upper exhaust outlet. Hence, it is essential to evaluate the shielding ability of the shielding house. The effective dose rate distributions around a transportable storage cask are obtained by ordinary Monte Carlo Calculations with the MCNPX code. The cross section library used for this analysis is JENDL-3.3 for neutrons and EPDL97 for secondary gamma-ray. The storage cask is stored in the shielding house which has sufficient multi layer structure. In this case, the gamma-ray from fission product and ^{60}Co activation is more shielded compared with neutron or secondary gamma-ray from neutron. Therefore, we are concerned here only with neutron and the secondary gamma-ray in the present study.

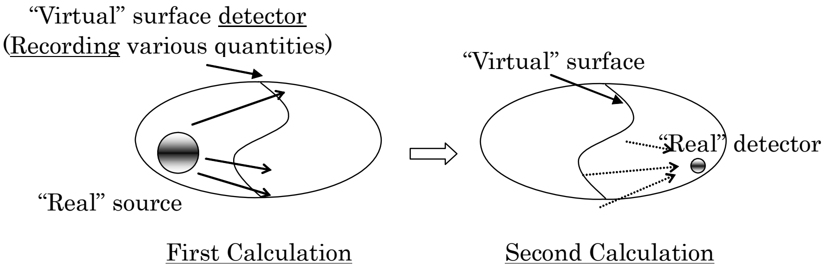


Figure 1.Schematic diagram of Monte Carlo Coupling Technique

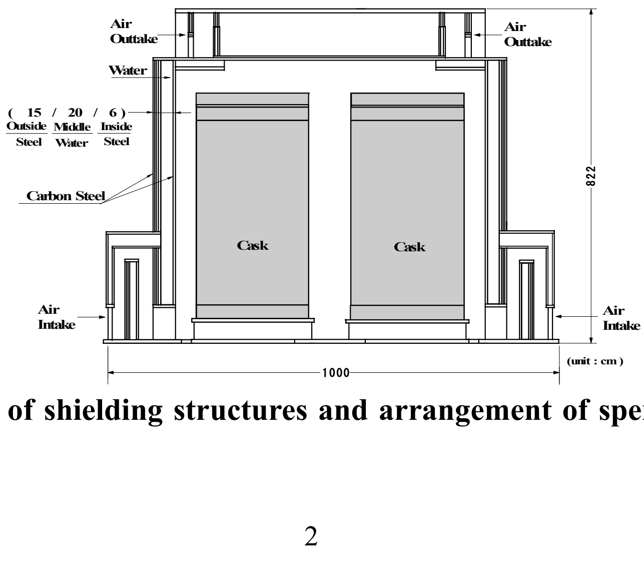
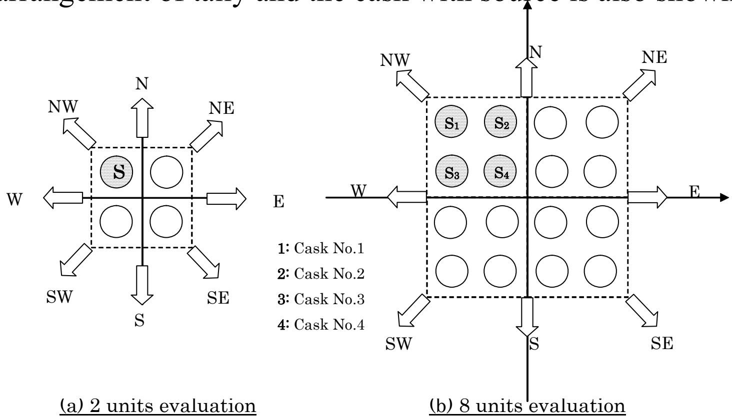


Figure 2.Overall view of shielding structures and arrangement of spent-fuel storage casks in a shielding house

THE CALCULATION METHOD OF “SHADE COEFFICIENT”

To formulate the “shade effect” index (shade coefficient) among storage casks, the effective dose rate per one cask source is calculated at the dose reference points on the line of eight directions as shown in Fig.3. The arrangement of tally and the cask with source is also shown in Fig.3.



(a) 2 units evaluation (b) 8 units evaluation

Figure 3.The arrangement of casks with source and evaluation direction of shielding performance for the storage facility

Computational results in the case of 2units shielding houses

[Neutron]

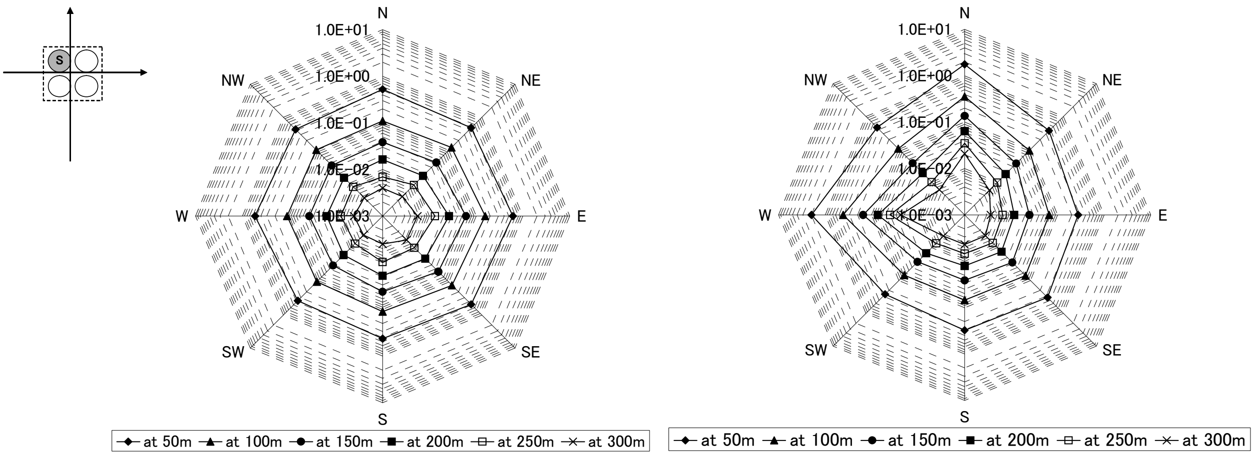
As shown in Fig.4 (a), the neutron emitted from the cask with source is hardly reduced by other casks. Hence, shade effect affects the neutron little.

[Secondary gamma-ray]

As shown in Fig.4 (b), the secondary gamma-ray emitted from the cask with source is reduced by other casks, particularly in the direction of E, SE and S; for example, the effective dose rate of direction-W is about six times as large as that of source direction-S.

[The distance from shielding house]

As shown in Fig.4, it is obvious that the tendency of the dose rate distribution hardly change according to the distance from the center of the shielding house. Thus the evaluation for 8 units shielding house is performed only at 100m from the center of shielding house.



(a) Neutron (b) Secondary gamma-ray

Figure 4.Effective dose distributions around the shielding house composed of 2 units (unit: μSv/year/1cask, Evaluated by PD)

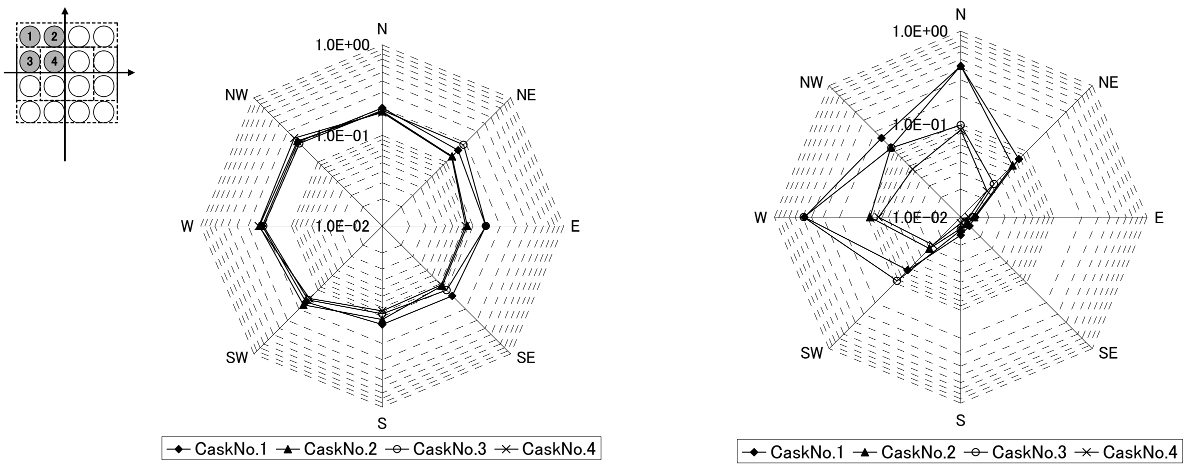
Computational results in the case of 8 units shielding houses

[Neutron]

As shown in Fig.5 (a), the dose difference depending on the direction N, NW, W is very little because only one cask shades the particle from the radiation source. Hence, the tendency of neutron dose is similar to that of the 2 unit-shielding houses. The increase of shielding cask contributes to the neutron scattering, particularly around direction-SE. However, the dose difference caused by neutron scattering cannot be calculated by shade effect. This is because the concept of shade effect is based on the attenuating process through obstacles.

[Secondary gamma-ray]

As shown in Fig.5 (b), there is significant difference between the direction near the radiation source (direction-SW/W/NW/N/NE) and the direction distant from radiation source (direction-E/SE/S). As for direction-E/SE/S, the number of cask which shade radiation source is 2 or more, thus secondary gamma-ray from radiation source is more shielded compared with the direction near the radiation source.



(a) Neutron (b) Secondary gamma-ray

Figure 5.Effective dose distributions around the shielding house composed of 8 units (unit: μSv/year/1cask, Evaluated by PD)

THE ESTABLISHMENT FOR THE USE OF A CONCISE ASSESSMENT

The skyshine dose from the upper exhaust outlet does not relate shade effect among casks, so the dose is calculated by following methodology:

1. The particle track that crosses outer surface of the cask is recorded by SSW option of MCNPX code.
2. Source particles generate from the outer surface of the cask. At the same time, the source particle through the upper exhaust outlet or upper outer surface of the shielding house is recorded by SSW option.
3. Source particles generate from the upper exhaust outlet or upper outer surface of the shielding house.
4. The effective dose rate at the facility boundary is evaluated.

In this study, the track of source particles is specified by using source particle write/read repeatedly as the SSW/SSR-CTR Repeating Method. This method is useful for this examination object such as bulk shielding problems and distant tallies.

SHADE COEFFICIENT AMONG CASKS

As for neutron, the interaction with casks is scattering process rather than attenuating process. As above, the dose difference caused by neutron scattering cannot be calculated by "shade effect" index. Hence, the effective dose rate of neutron is used in the case of 2 unit shielding house.

The shade coefficient for secondary gamma-ray is defined as follows:

$$C_{\text{shade}}|_{n=1}=(D\text{-}D_{\text{s}})/D,\ C_{\text{shade}}|_{n>1}=(C_{\text{shade}}|_{n=1})^{n}$$

Where

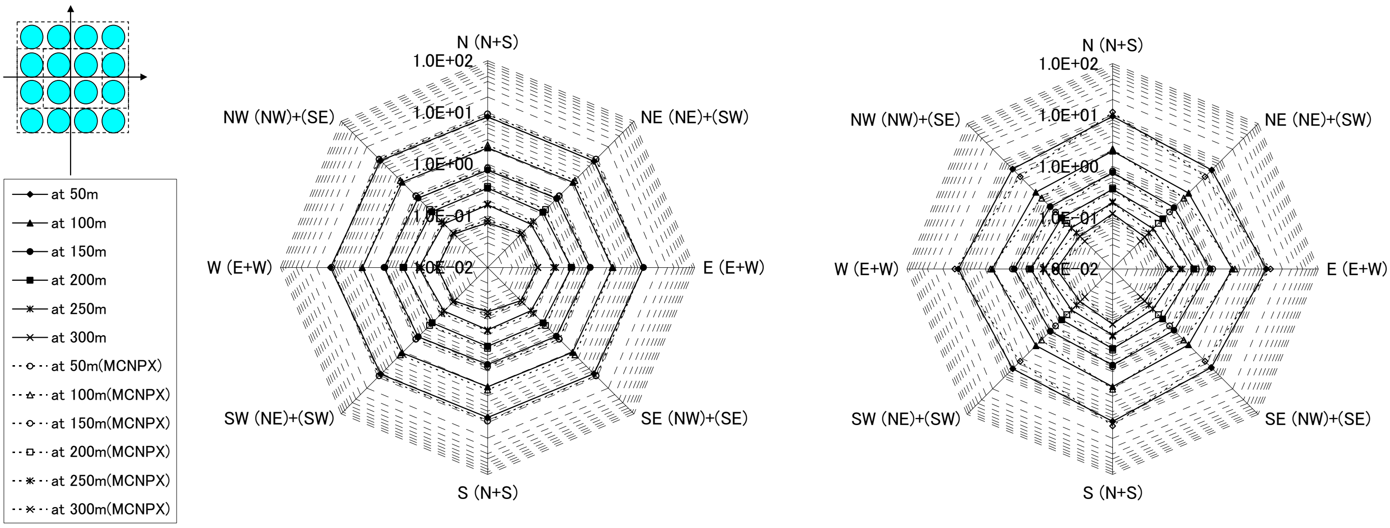
C_{shade}: shade coefficient

n: the number of casks acting as an obstacle when the particles generated from the radiation source come out of the shielding house

D: the effective dose rate without obstacles,

D_{s}: the effective dose rate when one cask exists in the direction of moving particles.

The shade coefficient of secondary gamma-ray among casks for the 2-unit shielding house is shown in Table 1.



(a) Neutron (b) Secondary gamma-ray

Figure 5.Comparison of calculated effective dose distribution around the shielding house composed of 8 units between with simplified calculation method by shade effect rate and with MCNPX (unit:µSv/year/16casks)

Table 1. Summary of shade effect rate for 2 units shielding house

Distance from the center of shielding house (m)		Shade effect rate (%)*							
Direction(-)		N	NE	E	SE	S	SW	W	NW
Secondary gamma-ray	50	No effect	82.5	28.1	85.9	No effect	20.4	No effect	43.3
	100		80.9	32.4	84.1		12.5		35.0
	150		81.0	25.3	83.8		2.65		28.4
	200		80.4	29.9	84.2		6.88		21.8
	250		80.5	28.9	83.6		3.43		27.3
	300		79.4	27.6	84.8		4.45		25.1

*** Shade effect coefficient is calculated without skyshine contribution.**

The calculated results with simplified method based on shade effect among casks are shown in Fig.5. The neutron contribution is consistent with the result of MCNP calculation without respect to shade effect. Secondary gamma-ray can be easily determined depending on “shade effect” index.

MCNP-ANISN_W – SIMPLE NEUTRON TRANSPORT CODE

A simple neutron transport code, MCNP-ANISN_W, which incorporates one-dimensional transport code, ANISN, into Monte-Carlo code, MCNP, is developed. The flowchart of the MCNP-ANISN_W calculation is shown in Fig.6. The features of this code are as follows:

Radiation source determination

Particle generation point is stochastically decided by the source region described in MCNP input. Particle energy is stochastically decided by the distribution of source spectrum described in MCNP input.

Distance between source and detector

The distance of point detector from the particle generation point in the source region is calculated by geometry data described in MCNP input.

Modeling

The cell boundary of geometry data described in MCNP input is determined as ANISN mesh boundary, material information and atom density in cell is computed and saved in the file.

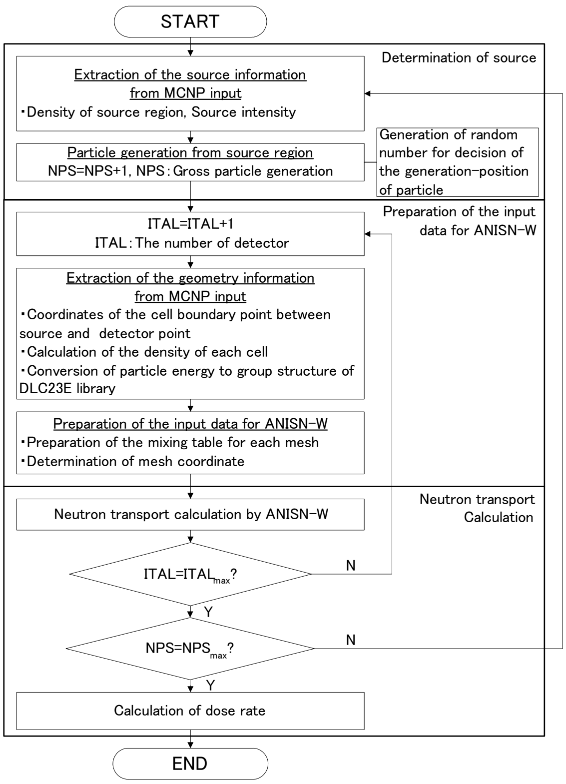


Figure 6.Flowchart of the MCNP-ANISN_W calculation

Benchmark calculation for MCNP-ANISN_W

This benchmark problem is defined the Aluminum sphere with a 9-cm radius. The neutron source which has the Watt fission spectrum of ^{235}U is considered at the center of the sphere with a 1-cm radius. The calculation is performed by means of some differential models for Sn calculation. The result of this calculation is shown in Fig.7. In particular, the result of weighted difference model is consistent with the MCNPX result above 1.0MeV. Therefore, weighted difference model could be used for all of the calculation.

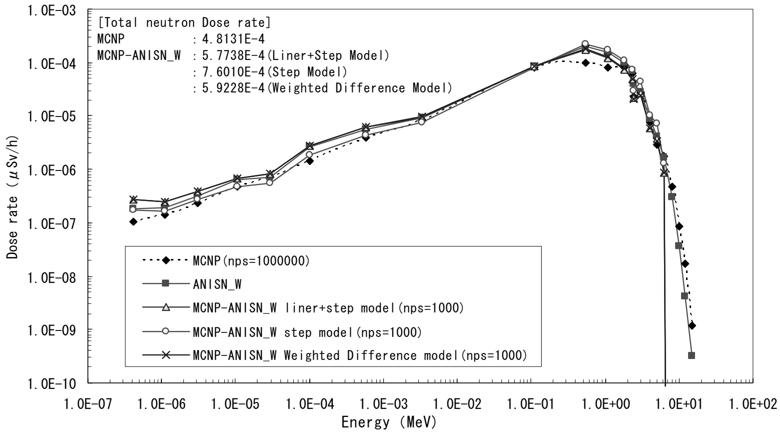
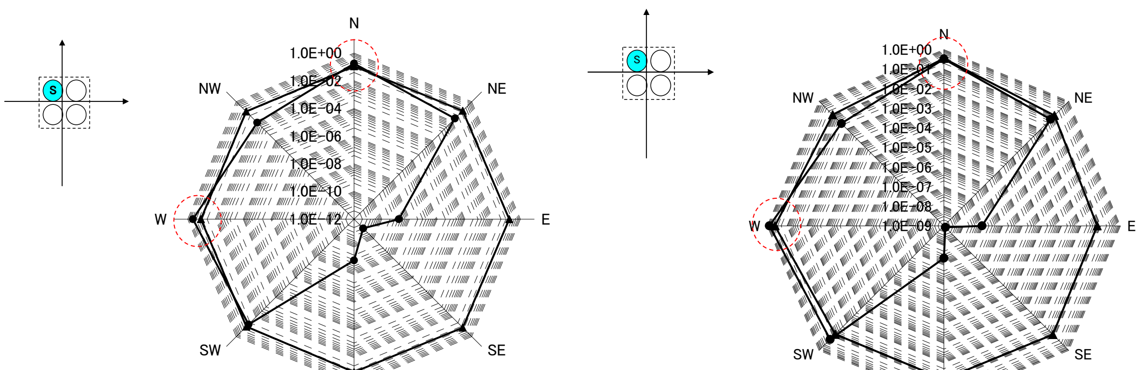


Figure 7.Comparison with MCNP-ANISN_W calculated results of liner+step/step/ step models

<u>The calculation results of the shielding house with MCNP-ANISN_W</u>

Owing to the one-dimensional code, the influence of neutron diffraction cannot be investigated, but the neutron behavior of direct radiation can be calculated by MCNP-ANISN_W. Contribution of direct radiation calculated with MCNP-ANISN_W code is shown in Fig.8. In the case of the effective dose rate for direction-W/NW/N, dose rate is approximately consistent with MCNPX result because of no obstacles in the neutron direction.



(a) Neutron (b) Secondary gamma-ray

Figure 8.Comparison with the calculated results of MCNP-ANISN_W and MCNPX (unit:µSv/year/1cask at 100m)

<u>The calculation results of the skyshine dose for the shielding house</u>

SHINE-III is the program which interpolates the distribution of the skyshine dose from a monochromatic neutron with the multi-group Monte Carlo code which took reflection by the ground into consideration. Contribution of skyshine calculated with SHINE-III is shown in Fig.9. Figure 9. also contains the contribution the direct radiation calculated with MCNP-ANISN_W. Compared with the results of MCNPX, total neutron dose rate calculated with MCNP-

ANISN_W and SHINE-III is overestimated. Hence, the results of SHINE-III are conservative. Therefore, sum total of neutron and secondary gamma-ray is consistent with the result of MCNPX.

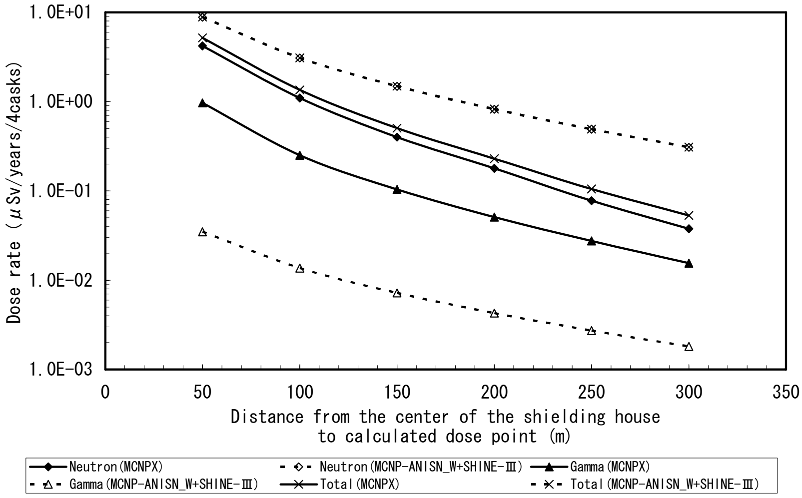


Figure 9.Comparison with the calculated skyshine-contribution of MCNPX and SHINE-III

CONCLUSIONS

Due to the calculation using the method of "shade effect", the effective dose rate for the interim facility stored 8 or 16 containers around the boundary was calculated and the results were consistent with that of Monte-Carlo calculation. A simplified calculation method based on shade effect among casks and MCNP-ANISN_W was developed. The results in the case of modular shielding house indicate that the equivalent results of Monte Carlo could be obtained with the code system in a short time. The simplified calculation method and the developed simplified code are very suitable for the system of many distributed source terms such as interim storage facility, furthermore, they could be applied to evaluate ships loading many casks..

ACKNOWLEDGMENTS

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