RADIATION SHIELDING ANALYSIS OF A SPENT FUEL TRANSPORT CASK WITH AN ACTUAL CONFIGURATION MODEL USING THE MONTE CARLO METHOD - COMPARISON WITH THE DISCRETE ORDINATES Sn METHOD -

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ABSTRACT

In order to demonstrate the features of Monte Carlo method, in comparison with the two-dimensional discrete ordinates Sn method, detailed modeling of the canister containing the fuel basket with 14 spent fuel assemblies, supplement shields located around the lower nozzles of the fuels, and the cooling fins attached on the cask body of the NFT-14P cask are performed using the Monte Carlo code MCNP 4C. Furthermore, the water level in the canister is assimilated into the present MCNP 4C calculations. For more precise modeling of the canister, the generating points of gamma rays and neutrons are simulated accurately from the fuel assemblies installed in it. The supplement shields located around the lower nozzles of the fuels are designed to be effective especially for the activation ^{60}Co gamma rays, and the cooling fins for gamma rays in particular. As predicated, compared with the DOT 3.5 calculations, the total dose-equivalent rates with the actual configurations are reduced to approximately 30 % at 1m from the upper side surface and 85 % at 1m from the lower side surface, respectively. Accordingly, the employment of detailed models for the Monte Carlo calculations is essential to accomplish more reasonable shielding design of a spent fuel transport cask and an interim storage cask.

Quality of the actual configuration model of the canister containing the fuel basket with 12 spent fuel assemblies has already been demonstrated by the Monte Carlo analysis with MCNP 4B, in comparison with the measured dose-equivalent rates around the TN-12A cask.

INTRODUCTION

Up to now, the two-dimensional discrete ordinates Sn method has been employed for radiation shielding analysis of spent fuel transport casks in Japan, and as an advantage contour maps of dose-equivalent rates around such casks can be described using the two-dimensional discrete ordinates Sn code DOT 3.5 (W. A. Rhoades, et al., 1973, and ccc-276, 1977) with DLC-23/Cask Library (ORNL-RSIC, 1973). However, the DOT 3.5 code has noticeable restrictions for the three-dimensional modeling of the shielding system in a cask, and so the canister containing the fuel basket and the cooling fins are homogenized, the supplemental shields located outside the lower nozzles are simplified and attached outside the fuel basket, and the trunnions are ignored in the DOT 3.5 calculations. As a result, the calculated dose equivalent rates tend to be overestimated considerably as compared with the measured data (Y. Momma, et al., 2000).

On the other hand, the Monte Carlo method is a very useful tool for solving a large class of radiation shielding problems. In contrast to the deterministic method, geometric complexity is a much less significant problem, and the Boltzman transport equation can be solved with any approximation. Before now, the gamma ray and neuron dose-equivalent rate distributions were measured around the TN-12A spent fuel transport cask and analyzed with the MCNP 4B code (J. F. Briesmeister Ed., 1997). In one case, the canister containing the fuel basket with 12 spent fuel assemblies was modeled three-dimensionally with the MCNP 4B code, and in the other case, two-dimensional homogenized model was used as employed in the Sn code DOT 3.5. By adopting the actual configuration with the Monte Carlo code, the gamma ray and neutron dose-equivalent rates were reduced to 67 % and 80 % respectively, on the TN-12A cask surface, as compared with the two-dimensional model of the Sn code. (K. Ueki, et al., 2000). Also, the homogenized model of the inside cavity and of the inside each assembly, and the model of the pin-by-pin representation of the spent fuel transport cask are analyzed with the MCNP code, and concluded that the homogenized model of the inside each assembly is generally satisfactory and the detailed modeling of the pin-by-pin representation is not necessary (G. Radulescu, et al., 2000). Accordingly, use of the Monte Carlo method would overcome the restrictions of the geometric configuration experienced with the deterministic methods; it would provide not only the best calculation results but also the reasonable shielding design of a transport cask and also an interim storage cask.

In the present study, the general purpose Monte Carlo code MCNP 4C (J. F. Briesmeister, Ed., 2000) with JENDL-3.2 library (T. Nakagawa, et al., 1995) has been employed and the effective dose rate distributions are obtained around a spent fuel transport cask NFT-14P. The NFT-14P cask is one of the most typical transport casks in Japan, and it is able to install 14 bundles of PWR (Pressurized Water Reactor) spent fuel assemblies. The main specifications are as follows; total weight is 115.0 tons, outer diameter is 2.6 m and height is 6.3 m, main structure is carbon steel, basket is composed of stainless steel both with and without boron, lead is used as gamma ray shield and resin named NS-4-FR for neutron shield, and it has cooling fins mode of stainless steel. Generally, following items are modeled in detail to produce the reliable assessment of the shielding properties of the cask in the MCNP 4C calculations; the canister containing the basket with spent fuel assemblies installed inside, supplement shields located around the lower part of the basket, and the cooling fins attached on the cask body. From the shielding point of view, the actual configuration model of the canister has effect on gamma rays and also the neutrons distinctly, supplement shields surrounding lower nozzles of the fuels are effective especially on the activation ^{60}Co gamma rays, and the cooling fins are effective on gamma rays in particular.

SHIELDING SYSTEM AND MODELING

At the first setout, the source intensity of the spent fuels is calculated by the ORIGEN2 code (A. G. Croff, 1980), and the effective multiplication factor, k_{eff}, of the cask, the KENO V.a (L. M. Petrie, et al., 1998) calculates with 14 PWR assemblies and k_{eff} of 0.66 is employed to obtain the neutron source intensity of the cask. The sources conditions, the gamma ray and the neutron source intensity of the spent fuels notified in the SAR (Safety Analysis Report) are summarized in TABLE II, III, and I respectively. Furthermore, as indicated in TABLE IV, numerous ^{60}Co activation gamma rays originating from ^{59}Co(n, γ) reactions in the stainless steel are produced in the fuel assemblies of the upper and lower nozzles, as well as in the upper plenums.

The axial-direction structures and the horizontal-plane structures around the lower nozzles of the NFT-14P modeled in the MCNP 4C calculations are shown in Fig. 1 and Fig. 2, respectively. The main shielding structures of the cask are lead and resin named NS-4-FR; 12.2 cm-thick lead is employed for gamma ray shielding and 16.8 cm-thick resin is for neutron shielding. As shown in Fig. 2, the NFT-14P cask can accommodate 14 PWR spent fuel assemblies in the basket made of stainless steel with boron. The cask is classified as wet type one, into which water is filled for cooling the decay heat and also shielding neutrons in the canister, and the supplement shields of 2.5 cm thick stainless steel are located to shield the activation ^{60}Co gamma rays from the lower nozzles at lower part of the fuel basket. Relatively high-energy photons of 1.17 and 1.33 MeV are produced from the ^{60}Co isotopes, and approximately 5.5-cm-thick steel is designated to reduce the dose-equivalent rate to 1/10. Also, the 77 cooling fins of stainless steel are attached on the cask body, and each fin is 0.8 cm thick, 16.0 cm in height, and distance between the fins is 4.6 cm (see Fig. 1). The dose points are also indicated by ①, and odd numbers are on the surface and even numbers are at 1 m from the cask surface In Fig's.1 and 2.

In particular, the canister containing the fuel basket with 14 spent fuel assemblies, the supplement shields located around the lower nozzles at lower part of the fuel basket, and the cooling fins around the cask body are modeled strictly to demonstrate the features of the Monte Carlo code MCNP 4C. By doing so, the generating points of gamma rays and neutrons from the fuel assemblies contained in the canister are simulated precisely, and also the shielding effects of the supplement shields around the lower nozzles against ^{60}Co gamma rays are evaluated with accuracy. Furthermore, the 77 cooling fins have some shielding effect for gamma rays. Meanwhile, all the structures including the cooling water in the canister are homogenized two-dimensionally, and the supplement shields are taken into account in the DOT 3.5 calculation but the shields are kept outside of the homogenized fuel region as a circular cylinder, and the cooling fins are treated as a void. Different assumptions between the DOT and the present MCNP 4C calculations are summarized in TABLE V. As described in INTRODUCTION, only the Monte Carlo analysis with the three-dimensional model of the fuel basket led to more excellent agreement with the measured neutron and gamma ray effective dose rates on and at 1m from the cask surface, as compared with the DOT 3.5 analysis.

TABLE I Source Conditions of Spent Fuels in the NFT-14P Cask

Reactor Tipe	PWR
Number of Assemblies	14
Initial Weight	470 (kg/Assembly)
Initial Enrichment	4.3 (%)
Burnup	44000 (MWD/MTU)
Specific Power	38.4 (MW/MTU)
Irradiation Time	1146 (days)
Cooling Time	630 (days)
k_{eff}	0.66

TABLE II Fission Products Gamma-Ray Source Intensity in the NFT-14P cask

Items	Gamma-Ray Source Intensity per Assembly				Gamma-Ray Source Intensity per Cask
	Top Edge (1/12)	Middle Part (10/12)	Bottom Edge (1/12)	Total	
Gamma Rays (photons/s)	2.335×10^{15}	2.789×10^{16}	2.335×10^{15}	3.236×10^{16}	4.530×10^{17}

TABLE III Neutron Source Intensity in the NFT-14P cask

Items	Neutron Source Intensity per Assembly				Neutron Source Intensity per Cask*
	Top Edge (1/12)	Middle Part (10/12)	Bottom Edge (1/12)	Total	
Neutrons (neutrons/s)	2.473×10^{7}	4.373×10^{8}	2.473×10^{7}	4.868×10^{8}	2.004×10^{10}

* Effective Multiplication Factor: K_{eff} =0.66.

TABLE IV Activation Gamma-Ray (^{60}Co) Source Intensity in the NFT-14P Cask

Activation Gamma-Ray (^{60}Co) Source Intensity per Cask			
Region	Upper Nozzles	Upper Plenums	Lower Nozzles
Content of Cobalt (g)	134	89.0	95.8
Irradiated Thermal Neutron Flux (n/cm^2/s)	1.7×10^{12}	2.4×10^{13}	6.7×10^{12}
Irradiation Time (days)	1146	1146	1146
Cooling Time (days)	650	650	650
^{60}Co Source Intensity (TBq)	23.2	217.3	65.3
^{60}Co Gamma Rays (p/s/cask)	4.64×10^{13}	4.346×10^{14}	1.306×10^{14}

Notes: Content of cobalt in the stainless steel is assumed to be 1.2 %.

RESULTS AND DISCUSSION

In order to obtain the reliable results in the MCNP 4C calculations, the different WWB's (Weight Window Bounds) are provided for neutrons and gamma rays respectively in each cell. The typical thickness of the cell is 0.5 cm for lead, 1.0 cm for steel, and 1.5 cm for the resin, and the WWB's for neutrons are 0.875m for lead, 0.825m for steel, and 0.725m for the resin, while the WWB's for gamma rays are 0.675m for lead, 0.625m for steel, and 0.850m for the resin, respectively. Subsequently, all the fsd's (fractional standards deviation) of the dose-equivalent rates on and at 1m from the cask surface are less than 10 %, which is evaluated as generally reliable for the Monte Carlo calculation. The detailed cell distributions to the axial direction and the horizontal direction around the lower fuel assemblies of the NFT-14P cask are shown in Fig. 1 and 2, respectively.

The dose-equivalent rates on the cask surface with and without the supplement shields are indicated in TABLE VI, and the shielding effect of the supplement shields against ^{60}Co activation gamma rays located around the lower nozzles are demonstrated. The most effective point of the supplement shields is at ⑦ on the lower side surface in Fig. 1, and the shielding effect is 0.406 (±0.0240), which means that the dose-equivalent rate is reduced to approximately 1/2.5 with the supplement shields in terms of the activation gamma rays. On the other hand, there is no effective shielding effect on the center side surface and also on the bottom surface, because the distance between the lower nozzles and the detector points are too long for the penetration of the gamma rays.

The maximum total dose-equivalent rates at 1 m from the cask surface are shown in Fig. 3, which also shows the comparison of total, neutron and gamma ray dose-equivalent rate distributions between the MCNP 4C and the DOT 3.5 calculations described in the safety analysis report. The gamma ray dose-equivalent rates consist of FP (Fission Products), ^{60}Co activation, and secondary gamma rays. The Japanese criteria of the cask dose-equivalent rates are on surface and at 1 m from the cask surface, and the criteria are 2 (mSv/h) on the surface and 100 (μSv/h) at 1m from the cask surface.

The maximum total dose-equivalent rates and fsd's at 1 m from the cask surface of the MCNP 4C calculations are as follows;

Lid :	21.912 μSv/h	(0.015)
Upper Side (Trunnion Side):	20.517 μSv/h	(0.015)
Center Body:	46.550 μSv/h	(0.024)
Lower Side (Trunnion Side):	60.426 μSv/h	(0.057)
Bottom:	22.985 μSv/h	(0.012)

The total dose-equivalent rates with the DOT 3.5 calculations are distributed between 30 % at the bottom and 70 % at lower side to the criteria, meanwhile the total doses with the MCNP 4C are reduced to between 20 % at the upper side and 60 % at the lower side to the criteria, and in comparison with the DOT 3.5 calculations, the MCNP results are approximately 30 % at the upper side surface and 85 % at the lower side surface, respectively. The main reduction factor of the total dose-equivalent rates at the upper side is the gamma ray, and neutron at the lower side in the MCNP calculations. However, the center of the fuel assemblies leads slightly to the bottom side, so that the total dose at the lower side is considerably high as compared with it at the upper side for both the MCNP calculations and also the DOT results in Fig. 3.

TABLE V Comparison of the Modeling Items between DOT 3.5 and MCNP Calculations for the NFT-14P Cask

Item	Present MCNP Calculation	DOT 3.5 Calculation in the SAR*
Fuel Basket and Fuel Assemblies	14 fuel assemblies are modeled one by one and installed in the basket.	Homogenized Uniformly
Suppremement Shields for the ^{60}Co Activation Gamma Ray	Stainless steel shields surrounded lower nozzles are modeled in detail.	Modified as a circular cylinder and set aside the homogenized fuel region.
Cooling Fins	77 Fins are modeled one by one.	Void

SAR* : Safety Analysis Report with DOT 3.5.

TABLE VI Shielding Effect of the Stainless Steel Supplement Shields Located around Lower Nozzles for ^{60}Co Activation Gamma Rays

^{60}Co Source Intensity in the Lower Nozzles: 1.306×10^{14} (photons/s/cask)

Detector Points in Fig. 1	A: Without Supplement SS[c] (μSv/h)	B: With Supplement SS[c] (μSv/h)	Shielding Effect B/A
⑤ On Center of Side Surface: 0 deg.	1.094×10^{-2} (0.0310)[a]	9.586×10^{-3} (0.0276)[a]	0.867 (±0.0360)[b]
On Center of Side Surface: 90 deg.	1.055×10^{-2} (0.0212)	1.013×10^{-2} (0.0400)	0.960 (±0.0453)
⑦ On Lower Part Surface: 0 deg.	23.336 (0.0337)	9.472 (0.0487)	0.406 (±0.0240)
On Lower Part Trunnion : 90 deg.	5.390 (0.0275)	3.522 (0.0311)	0.653 (±0.0271)

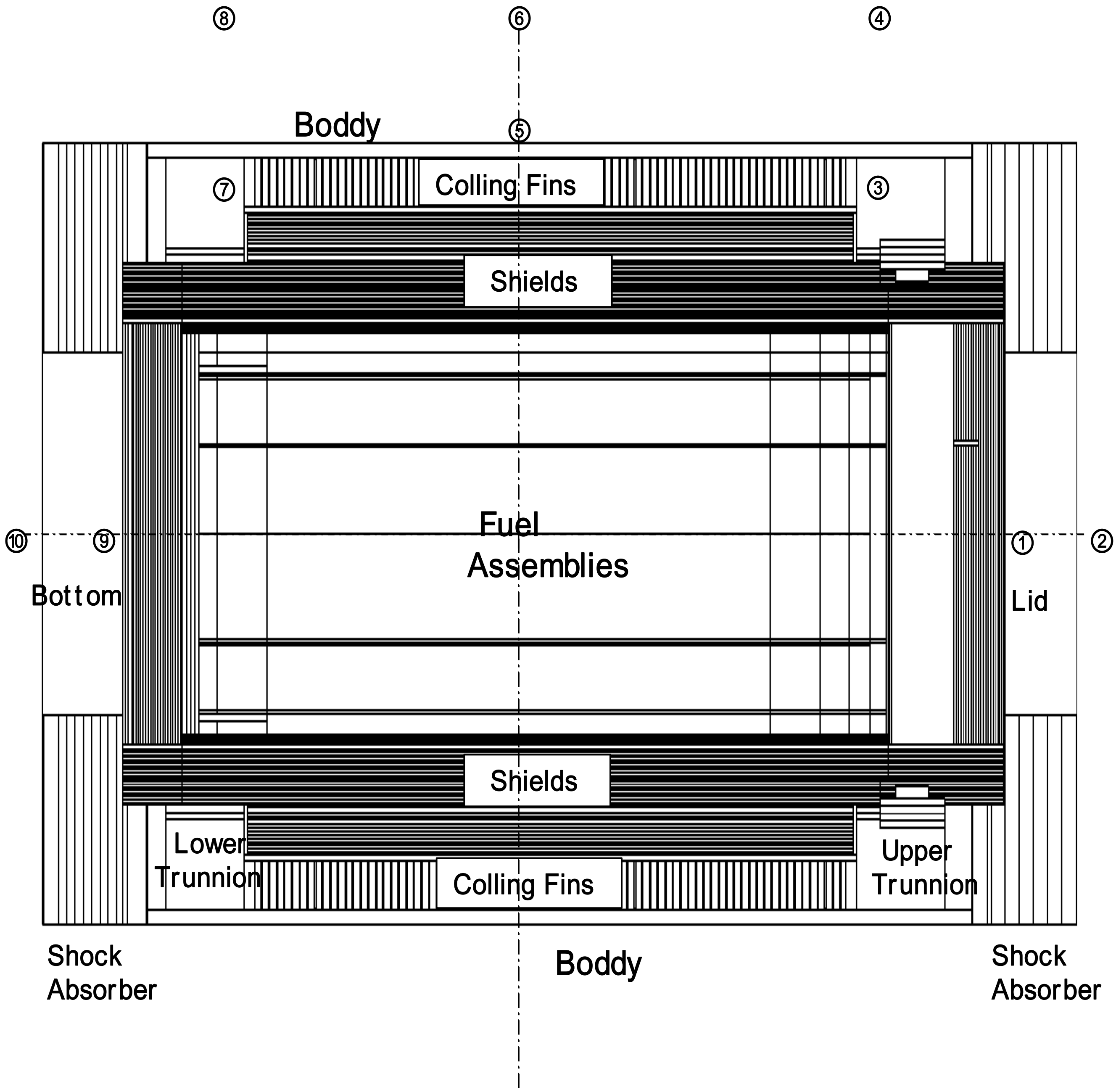

Fig. 1 MCNP Axial-Direction Model and Dose Points around the Body of the NFT-14P Cask.

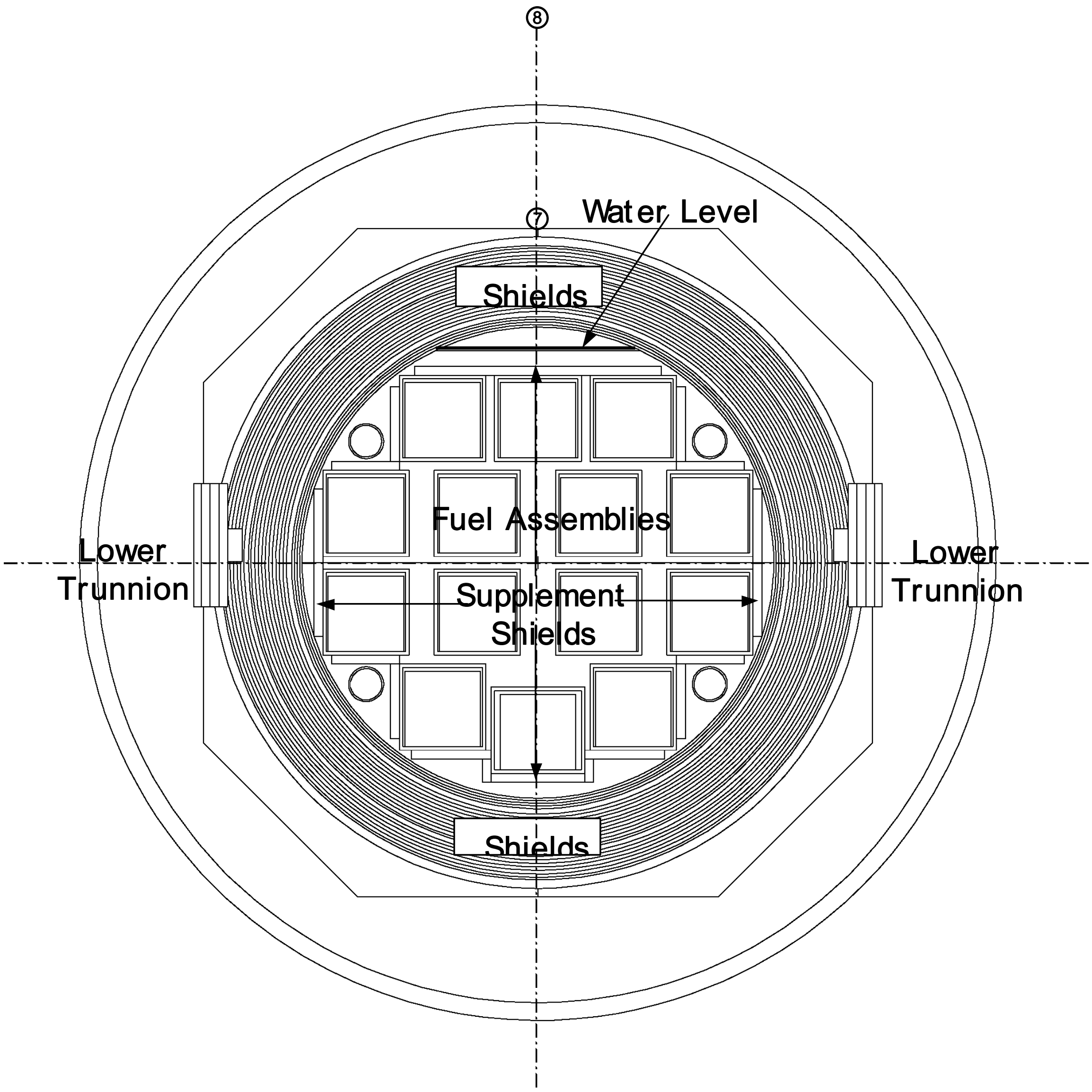

Fig. 2 MCNP Horizontal Plane Model of the Fuel Assemblies and the Supplement Shields around Lower Part of the Fuel Basket in the NFT-14P Cask, and Dose Points around Them.

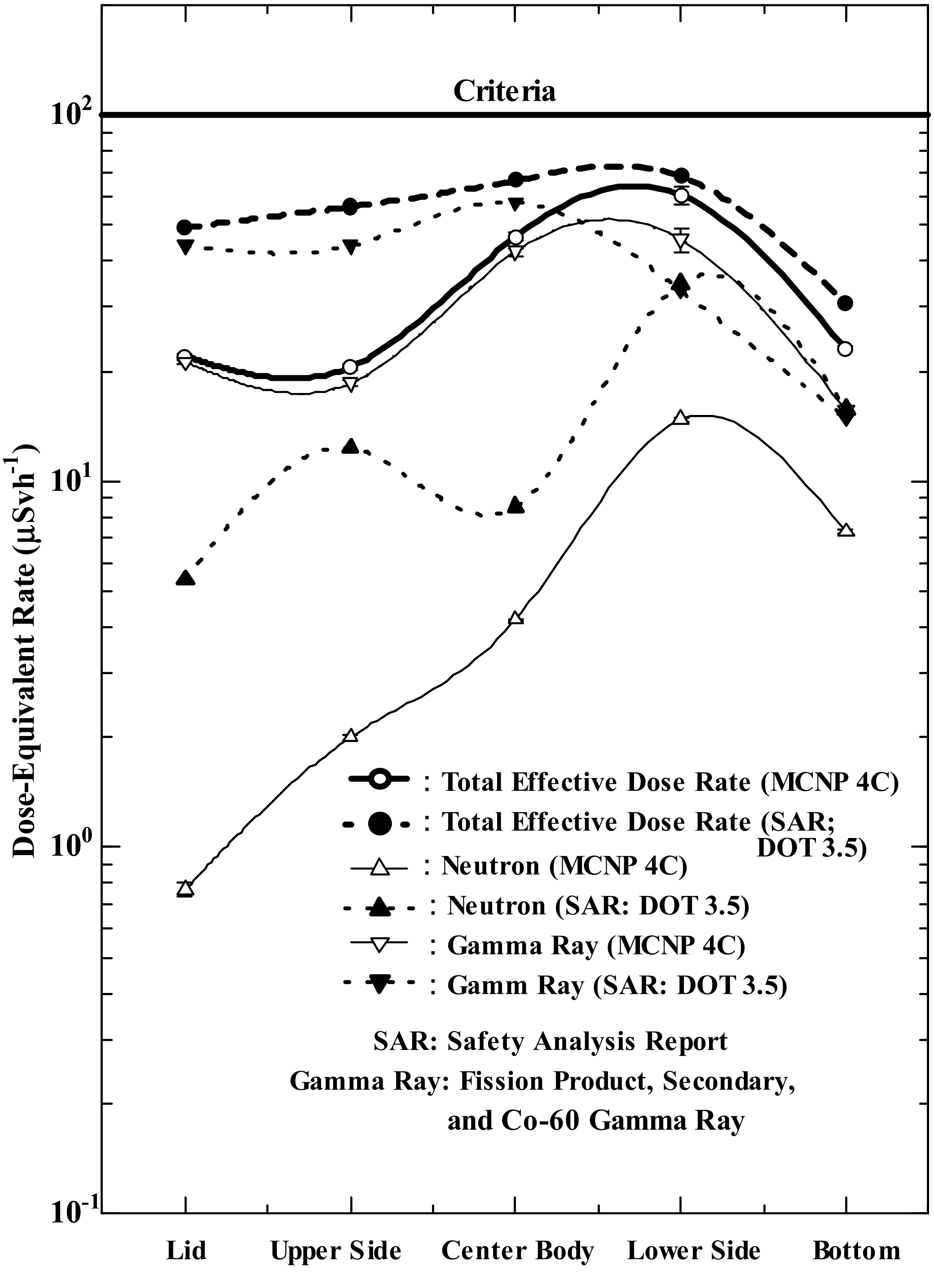

Fig. 3 Distributions of Effective Dose Rates at 1m from the NFT-14P Cask Surface.

CONCLUDING REMARKS

The following remarks can be made from the present analysis by using the Monte Carlo calculations with the actual configuration model of a spent fuel transport cask.

1. Best of all, following three items are modeled in detail in the present MCNP 4C calculations; the canister containing the fuel basket with 14 spent fuel assemblies, supplement shields located around the lower nozzles of the fuels, and cooling fins attached on the cask body of the NFT-14P cask.
2. By doing so, the generating points of gamma rays and neutrons are simulated precisely from the fuel assemblies contained in the canister, and also the shielding effects of the supplement shields around the lower nozzles against ^{60}Co gamma rays are evaluated with accuracy. On the other hand, when the fuel basket of stainless steel with the fuel assemblies is homogenized, the source generating points are distributed uniformly in the homogenized region. Accordingly, the source generation from the outer part of the homogenized canister leads the overestimation compared with the actual model of it. In consequence, the dose-equivalent rates of the DOT 3.5 calculations results in distinct overestimation in comparison with the MCNP 4C calculations, and the total dose-equivalent rates with the actual configurations are reduced to approximately 30 % at 1m from the upper side surface and 85 % at 1m from the lower side surface respectively, as compared with the DOT 3.5 calculations.
3. As expected, the shielding effect of the supplement shields located around the lower nozzles is quite remarkable on the lower part side surface; in comparison with no shields, the dose equivalent rate of the activation ^{60}Co gamma rays is reduced to 1/2.5 on the side surface.
4. Based on the results of the present MCNP 4C calculations, the total dose-equivalent rates are distributed between 20 % and 60 % of the Japanese criteria, and indicating that the Monte Carlo calculations are indispensable to overcome the excessive shields and also to enable reasonable shielding design of a spent fuel transport cask and an interim storage cask.
5. As a future issue, the shielding performance of a shielding house in which a quite large number of spent-fuel transportable storage casks are accommodated, the Surface Source Write (SSW) and Surface Source Read/Coordinate Transformation (SSR/CRT) code system of the MCNP (K. Ueki, et al., 2003) is able to evaluate it with actual configuration of the shielding system.

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