Evaluation of radiation shielding performance in sea transport of radioactive material by using simple calculation method

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ABSTRACT

A modified code system based on the point kernel method was developed to use in evaluation of shielding performance for maritime transport of radioactive material. For evaluation of shielding performance accurately in the case of accident, it is required to preciously model the structure of transport casks and shipping vessel, and source term. To achieve accurate modelling of the geometry and source term condition, we aimed to develop the code system by using equivalent information regarding structure and source term used in the Monte Carlo calculation code, MCNP. Therefore, adding an option to use point kernel method to the existing Monte Carlo code, MCNP4C, the code system was developed. To verify the developed code system, dose rate distribution in an exclusive shipping vessel to transport the low level radioactive wastes were calculated by the developed code and the calculated results were compared with measurements and Monte Carlo calculations. It was confirmed that the developed simple calculation method can obtain calculation results very quickly with enough accuracy comparing with the Monte Carlo calculation code MCNP4C.

1. INTRODUCTION

Various methods are used for calculation of dose distribution around transport casks and transport vessels as evaluation of shielding performance in sea transport of radioactive material according to an object and required accuracy. Widely used Monte Carlo calculation code, MCNP, is possible to apply to complicated geometry problem because the code can preciously model the geometry and can eliminate uncertainty due to the geometry modelling. While the Monte Carlo method has such an advantage, long calculation time is required for large scale problem to get adequate accuracy even if we use current high performance computing system because the method is based on the stochastic method. One- or two-dimensional deterministic code has an advantage that calculation results can be obtained relatively short time compared with the Monte Carlo method while uncertainty arising from geometry modelling should be considered. Therefore, the deterministic calculation codes are often used for parametric survey of shielding structure. The point-attenuation-kernel integration method (point kernel method), which is based on an analytical method, is also used for evaluation of shielding performance and shielding design. A code based on the point kernel method can obtain results very quickly and is applicable to relatively complex system with combinatorial geometry method. Considering the advantage of the point kernel code, the point kernel code is suitable to support emergency response of competent authority in case of the accident during sea transportation of radioactive material. In this study, a modified code system based on the point kernel method was developed to use in evaluation of shielding performance for maritime transport of radioactive material. In this paper, overview of the code system and results of verification of the code system are described.

For accurate evaluation of shielding performance in the case of accident, it is required to preciously model structure of the transport casks and shipping vessel, and sour term. To achieve accurate modelling of the geometry and source term condition, we aimed to develop the code system by using equivalent information regarding structure and source term used in the Monte Carlo calculation code. Therefore, the code system was developed by adding an option to use point kernel method to the existing Monte Carlo code, MCNP.

2. POINT KERNEL INTEGRATION OPTION IN MONTE CARLO CALCULATION CODE

A Monte Carlo calculation code MCNP4C [1] is used to evaluate dose rates around transport packages and shipping vessels in a supporting system for emergency response to maritime transport accidents involving radioactive material developed by National Maritime Research Institute (NMRI) by consignment of Ministry of Land, Infrastructure and Transport, Japan. NMRI has been conducting modelling of various types of packages and shipping vessels for calculation of shielding performance by MCNP code. For effective development of a simple calculation method, it is better to use the same input data for the calculation code in the supporting system.

In this study, the point kernel integration method was implemented to shielding calculation code without modification of the source routine in the MCNP. Gamma-ray energy and coordinate where the gamma-ray produced in a source region are sampled from information provided by source information in input data and initial weight of the gamma-ray is determined. The initial weight of the gamma-ray represents probabilistic integration factor rather than deterministic integration factor regarding source term.

Dose rate contribution $R(r)$ by the point kernel to some point detector tally determined from probabilistic sampling of source is

$$R(r) = W \cdot F \cdot B \cdot R^0 , \tag{Eq. 1}$$

where W is initial weight of gamma-ray, F is uncolided flux, B is buildup factor, and R^0 is flux-to-dose conversion factor, respectively. The uncolided flux, F , can be determined by

$$F = \frac{\exp(-L)}{4\pi \cdot r^2} \quad . \tag{Eq. 2}$$

In Eq. 2, r is distance between a probabilistically sampled point in the source region and a point detector tally, and L is number of mean free paths for distance r .

Function for the point detector tally in the MCNP code calculates the uncolided flux, F , by calculating number of mean free paths between a source point to the detector point as shown in Fig.1. In calculation of number of mean free paths, total cross section of gamma-ray of the cell material is used instead of the mass absorption coefficient. To calculate dose contribution $R(r)$ by the point kernel method in the MCNP, the functions to evaluate number of mean free paths, the buildup factor and the flux-to-dose conversion factor are newly required. To calculate number of mean free paths, path length of each passed cells from source point to detector point and the mass absorption coefficients are used.

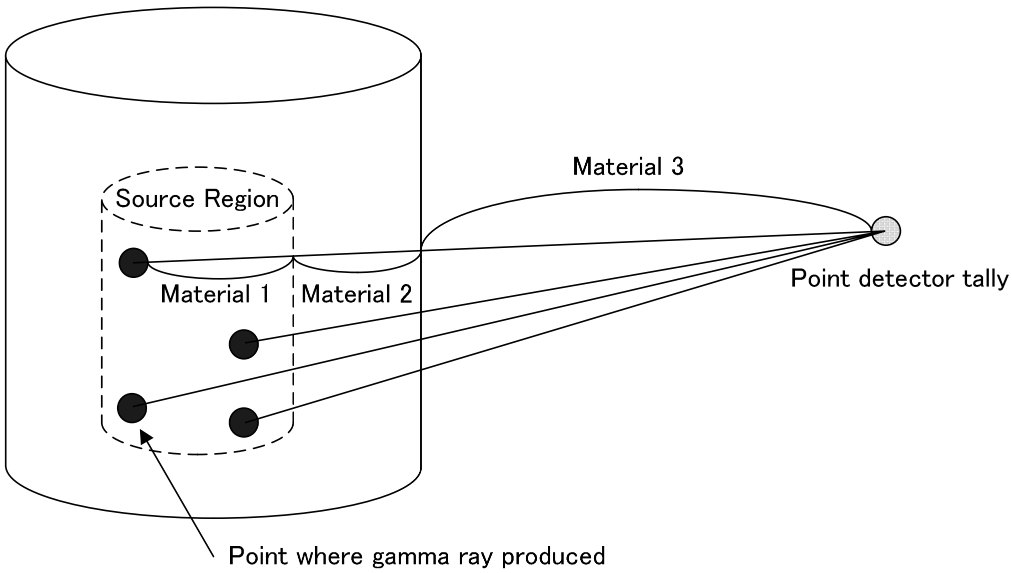

Fig.1 Relationship between points where gamma-ray produced and point detector is located.

For evaluation of the buildup factor, geometric propagation formula (GP form) [2] can be used. Series of QAD code system are well known as a calculation code adopting GP method for evaluation of buildup factor. In the present study, QAD-CGGP2R code [3,4], that the effective dose rate conversion factor, the air kerma-rate coefficient are implemented in the code, was adopted. In the supporting system for emergency response to maritime transport accidents involving radioactive material, the MCNP4C code is used as a Monte Carlo calculation code. In this study, functions implemented in the QAD-CGGP2R code to calculate number of mean free paths, buildup factor and flux-to-dose conversion factor are added to the MCNP4C code.

3. DEVELOPMENT OF INTERFACE FUNCTIONS BETWEEN MCNP4C3 AND QAD-CGGP2R

For implementation of functions in QAD-CGGP2R to MCNP4C, interface functions between both codes are developed.

In the MCNP code, the following calculation procedures are controlled:
(1) determination of produced point of gamma-ray in the source region by means of probabilistic method,
(2) determination of gamma-ray energy for given energy spectrum by means of probabilistic method,
(3) determination of path length between source point to detector point by using a function of point detector tally,
(4) storing crossing coordinates that path between source point and detector, and cell boundary crosses, and material information of each boundary.

Above information is transferred to functions to calculate number of mean free paths, the buildup factor and the flux-to-dose conversion factor. Using these information QAD functions evaluate number of mean free paths, uncolided flux, F , the buildup factor on the basis of GP method, and contribution to dose rate, and then transfer calculated results to the MCNP code.

4. BENCHMARK CALCULATION

To verify basic function of the developed simple calculation code, an effective dose rate was evaluated for simple system using the developed code, MCNP4C, and QAD-CGGP2R. A cubic whose one side is 14cm, inside of the region is filled with hypothetical material containing uranium and lithium, is assumed as shown in Fig.2. An effective dose rate was evaluated at a detector point, (x, y, z) = (8.0, 0.0, 0.0), by the developed code, MCNP4C, and QAD-CGGP2R. Gamma-ray source is generated in a cubic whose one side is 0.2 cm, located in center of the system. The source region consists of uranium and hydrogen. Typical gamma-ray spectrum calculated for spent fuel using ORIGEN code [5] was adopted and 18 group spectrum was generated.

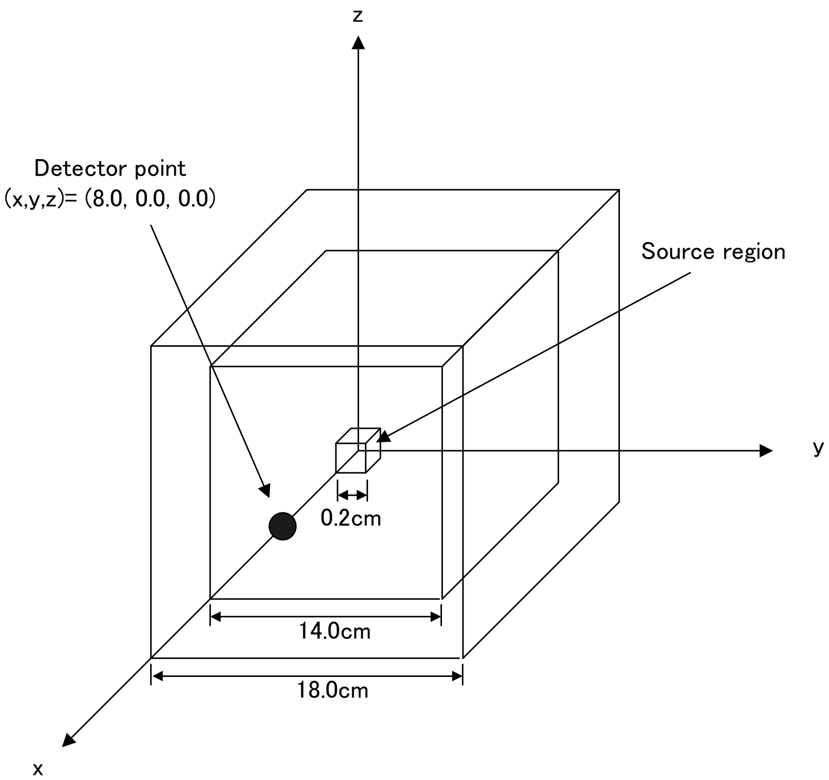

Fig. 2. Geometry of benchmark calculation.

Calculated effective dose for each gamma-ray energy group is shown in Fig. 3. The effective dose calculated by the developed simplified code and QAD-CGGP2R code is in good agreement. Spectrum of the effective dose for each energy group calculated by the MCNP4C code shifts in lower energy groups compared with QAD calculation because lower energy gamma-ray component is increased due to multiple scattering of gamma-rays in the system. Good agreement between the developed simplified code and MCNP4C can be found in high energy region. Calculated effective dose rate at the detector point by MCNP4C and QAD calculation agreed within 10 %. Results of the benchmark calculations indicate that implementation of point kernel integration method to MCNP4C was properly carried out.

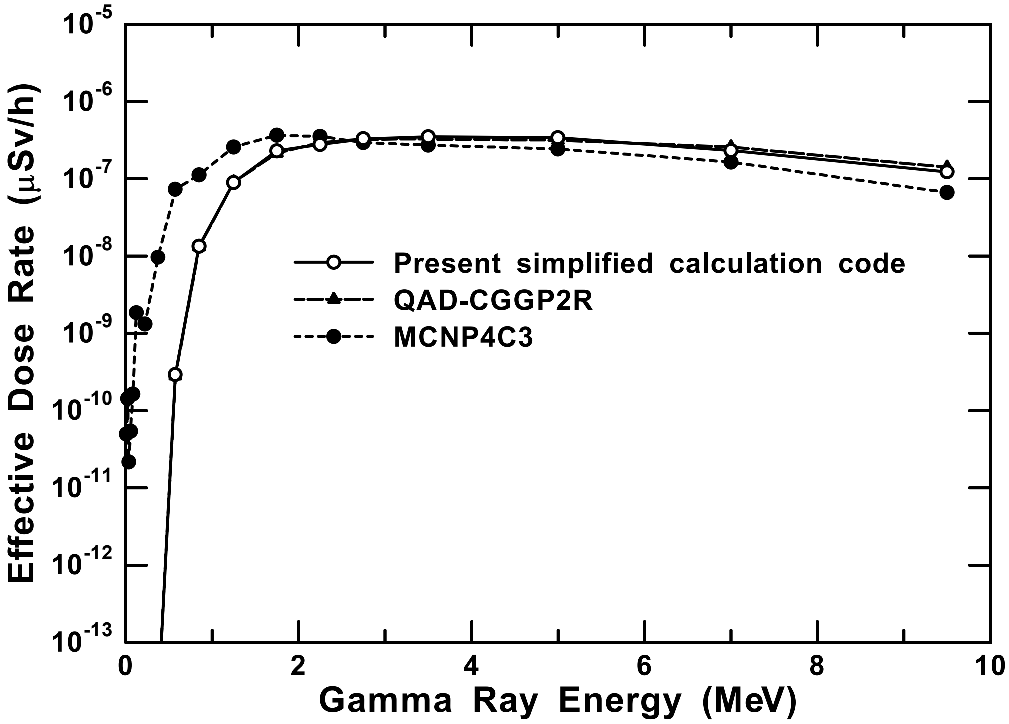

Fig.3 Effective dose for each gamma-ray energy group.

5. APPLICATION OF SIMPLIFIED CALCULATION CODE TO RADIOACTIVE MATERIAL SHIPPING VESSEL

As an example of application of the developed simplified calculation code to radioactive material shipping vessel, evaluation of dose distribution was carried out for low level radioactive waste shipping vessel.

5.1 MEASUREMENT OF DOSE RATE EQUIVALENT INBOARD SHIPPING VESSEL

Detailed measurement for dose equivalent rates above hatch cover of each hold and the accommodation area was carried out in low level radioactive waste (LLW) shipping vessel [6]. As shown in Fig.4, 360 containers were loaded in No. 2 to 7 hold of the LLW shipping vessel. Eight LLW drums were loaded on a container and 60 containers were loaded on each hold. Volume of each LLW drum is 200 litter and LLW drums are solidified with concrete. Most of the gamma ray source of the LLW is ^{137}Cs and ^{60}Co. Gamma-rays from ^{58}Co can be negligible because amount of ^{58}Co is 10% of ^{60}Co at the time of solidification of the LLW drum and shipping of the LLW drums are usually carried out 5 years later after solidification, that is very long compared with its half life, 70.8 days. Table 1 shows gamma-ray strength of LLW loaded on each hold when dose measurement was carried out.

Dose equivalent rate distribution was measured by the NaI(TI) scintillation survey meter. Accuracy of the measured dose equivalent was 15 %. Measured points in the LLW shipping vessel is shown in Fig.4 and measured results are shown in Table 2.

5.2 BENCHMARK ANALYSIS

Source region was homogeneously modelled for every container which contained 8 LLW drums. Density of the source region is 1.69 g/cm^3. Source strength was given for each container so that total activity in each hold is equivalent to that in Table 1. For benchmark analysis, the developed simplified calculation code and MCNP4C which is implemented to the supporting system for emergency response to maritime transport accidents involving

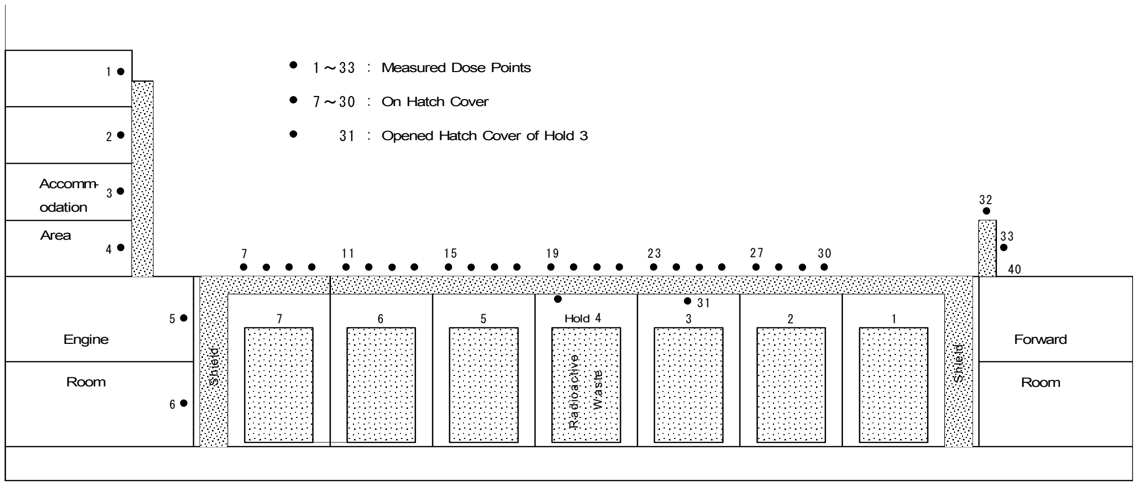

Fig. 4 Schematic view of LLW shipping vessel.

Table 1. Gamma ray source strength of LLW in each hold.

Hold No.	^{137}Cs (photons/s)	^{60}Co (photons/s)
2	9.62×10^{10}	4.2×10^{10}
3	8.26×10^{10}	4.2×10^{10}
4	7.82×10^{10}	4.1×10^{10}
5	5.30×10^{10}	4.6×10^{10}
6	2.74×10^{10}	6.2×10^{10}
7	2.22×10^{10}	4.4×10^{10}

Table 2. Measure dose rate equivalent.

Measured point	Measured dose rate equivalent (μSv/h)	Measured point	Measured dose rate equivalent (μSv/h)
1	0.05 ± 0.0075	18	0.92 ± 0.138
2	0.045 ± 0.00675	19	0.63 ± 0.0945
3	0.045 ± 0.00675	20	0.96 ± 0.144
4	0.025 ± 0.00375	21	0.78 ± 0.117
5	0.025 ± 0.00375	22	0.66 ± 0.099
6	0.02 ± 0.003	23	0.40 ± 0.06
7	0.50 ± 0.075	24	0.57 ± 0.085
8	0.82 ± 0.123	25	1.00 ± 0.15
9	0.77 ± 0.1155	26	0.91 ± 0.1365
10	0.63 ± 0.0945	27	0.39 ± 0.0585
11	0.59 ± 0.0885	28	0.58 ± 0.087
12	1.09 ± 0.1635	29	0.83 ± 0.1245
13	0.81 ± 0.1215	30	0.75 ± 0.1125
14	0.58 ± 0.087	31	19.7 ± 2.955
15	0.58 ± 0.087	32	0.04 ± 0.006
16	1.10 ± 0.165	33	0.01 ± 0.0015
17	1.17 ± 0.1755		

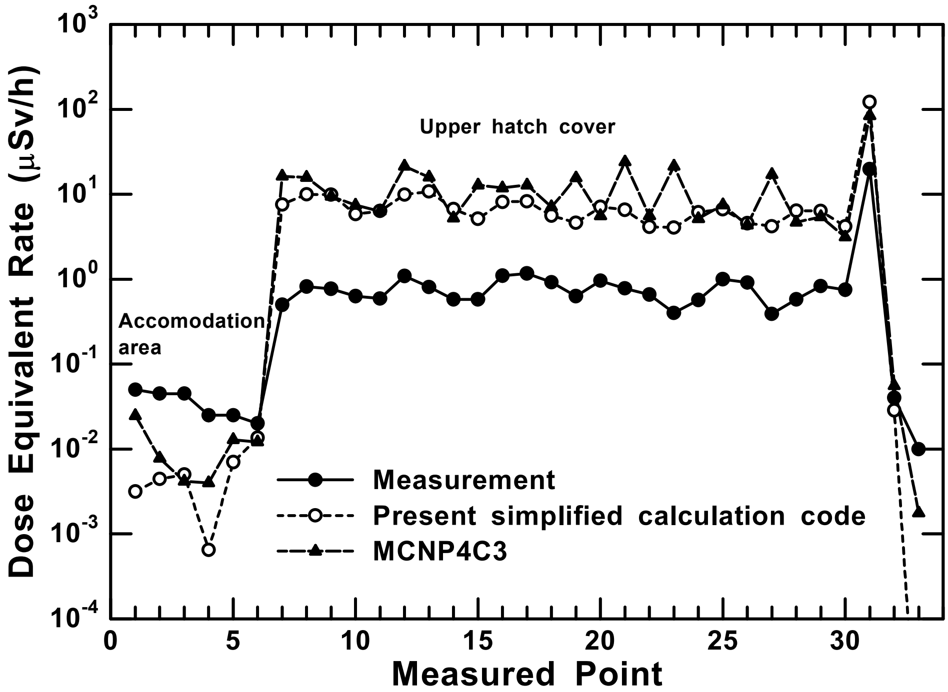

Fig. 5 Results of benchmark analysis for LLW shipping vessel.

radioactive material. Comparison between measured results and calculation is shown in Fig. 5. Vertical axis represents detector points and the numbers correspond to ones in Fig. 4 and Table 2.

Tendency of the results by the developed simplified calculation code and the MCNP4C code are in good agreement. Though the MCNP results overestimated considerably at several points, further detailed analyses are required for solve this discrepancy because the fractional standard deviation of those results were very large and the results are not reliable.

Comparing calculated results by simplified code with measurement, dose equivalents above the hatch cover are conservative evaluation one order of magnitude and tendency is almost same. Reasons of overestimation above the hatch cover are as follows: conservative values are used for source term, and gamma-ray is easier to penetrate because source region was homogenized and density of the source region is lower than that of concrete. The calculated results are relatively in good agreement in the engine room; the calculated results by the simplified calculation code are underestimated compared with measured data and tendency is different from results by the MCNP4C code. Because the point kernel integration method can not take into account effect of the sky shine, dose rates in the accommodation area would be underestimated.

Calculation speed is one of importance characteristics for a code used in the emergency response. Comparison of calculation speed of the developed simplified calculation code and MCNP4C is shown in Table 3. Calculation was carried out for the LLW shipping vessel setting two detectors. Computer used for calculation is a Linux PC with Pentium 4, 1.7 GHz, CPU. The present result clearly shows that calculation speed of the simplified calculation code is very fast.

Table 3. Comparison of calculation speed.

	Simplified calculation code	MCNP4C
Number of particles generated	100,000	5,000,000
Calculation time (min.)	3.89	3738.91

Though the developed simplified calculation code has disadvantage that the code can not take into account sky shine effect, it is suitable for emergency response, which require quick evaluation, because the code system can evaluated dose distribution for many points in very short time with good enough accuracy for the emergency response. Effect of sky shine would be taken into account using data base which can be prepared from detailed Monte Carlo calculation.

6. CONCLUSION

A simplified calculation method based on the point kernel integration method was implemented to the shielding calculation code system which constitutes the supporting system for emergency response to maritime transport accidents involving radioactive material developed by NMRI aiming support of emergency response by Ministry of Land, Infrastructure and Transport. The simplified code system was developed by adding functions implemented in the QAD-CGGP2R code to calculate number of mean free paths, the buildup factor and the flux-to-dose conversion factor are added to the MCNP4C code. To verify validity of the developed code, measured data of dose rates distribution in the LLW shipping vessel was analyzed by the developed code. The results of the benchmark analyses indicate that the present code system can obtain equivalent results to that of Monte Carlo calculation in short time. The developed simplified code system is very suitable for the emergency response that requires quick evaluation of shielding performance for relatively complex geometry.

ACKNOWLEDGMENT

This study was conducted under a contract research “Development of accident assessment system for maritime transport of radioactive material” with Ministry of Land, Infrastructure and Transport, Japan.

REFERENCES

[1] J.F. Briesmeister, Editor, “MCNP™ – A General Monte Carlo N – Particle Transport Code, Version 4C”, LA-13709-M (2000).
[2] Y. Harima, Y. Sakamoto, S. Tanaka and M. Kawai, “Validity of the Geometric-Propagation Formula in Approximating Gamma-Ray Buildup Factors”, Nucl. Sci. Eng., 94, 24 (1986).
[3] Y.Sakamoto and S.Tanaka, "QAD-CGGP2 and G33-GP2: Revised Versions of QAD-CGGP and G33-GP. Codes with the Conversion Factors from Exposure to Ambient and Maximum Dose Equivalents", JAERI-M 90-110 (1990).
[4] Y. Sakamoto, A. Endo, S. Tsuda, F. Takahashi and Y. Yamaguchi, "Shielding Calculation Constants for Use in Effective Dose Evaluation for Photons, Neutrons and Bremsstrahlung from Beta-Ray", JAERI-Data/Code 2000-044 (2001) (In Japanese).
[5] A.G. Croff, “A User’s Manual for the ORIGEN2 Computer Code”, ORNL/TM-7175 (1980).
[6] K. Ueki, A. Ohashi and N. Nariyama, “Evaluation of Radiation Safety in Sea Transport of Low Level Wastes”, Papers of Ship Research Institute, 33 (4), 1 (1996) (In Japanese).