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ARTICLE

Introduction of "Guideline for Radiation Shielding Evaluation of Transport Casks by Monte Carlo Method" -sample calculation-2: NFT-14P wet type cask-

Tsutomu Matsumoto[a*], Shinya Ohishi [a] and Naotsugu Kikuchi [b]

[a]*OCL Corporation, 1-1-3, Shiba Daimon, Minato-ku, Tokyo, 105-0012, Japan;* [b]*Nuclear Fuel Transport, 1-1-3, Shiba Daimon, Minato-ku, Tokyo, 105-0012, Japan*

Introduction of "Guideline for Radiation Shielding Evaluation of Transport Casks by Monte Carlo Method" is planned in Japan by National Maritime Research Institute (NMRI) because so for Monte Carlo codes are not popular to use for these kind of shielding analysis. Target of this Guideline is transport casks loading high level radioactive materials such as spent fuel. With a view to preparing the backup data to support the Guideline, we perform the radiation shielding calculation by using MCNP code for NFT-14P wet type cask which is the representative spent fuel cask of Japan. The measurements of dose rate for this cask were performed in 2009 by NMRI. The measurement results for neutrons and gamma rays for the cask loaded 14 PWR spent fuels were shown. All calculations pass the relative deviation criteria. With regard to 10 statistical checks provided by the MCNP output, some articles are not satisfied with the criteria in this study. However, the calculated dose rate obtained by the point detector and the track length estimator are in good agreement. In addition, in comparison with the measurements, the axial distribution profiles of the dose rate are in good agreement. These trends represent that the calculated dose rate are acceptable as a reasonable analysis result. In many cases, the calculated dose rate is larger than the measured dose rate, and therefore the calculation procedure presented in the Guideline is conservative and useful for the purpose of the safety analysis.

Keywords:* *radioactive material transport cask; Monte Carlo method; safety analysis; the guideline; variance reduction; NFT-14P

1. Introduction

The MCNP code's calculation model in this study is a detailed three-dimensional model, which models the fuel basket and the trunnion as drawings[1]. The input data of the calculation is created generally in accordance with the Guideline which is the conservative procedure[2].

The source intensity is calculated by ORIGEN2.2-upj code[3]. Since the neutron source intensity of spent fuel assembly depends a great deal on the burn-up distribution in the axial direction, the burn-up distribution is taken into account in the source calculation.

For the variance reduction technique necessary for non analog Monte Carlo calculation, we employ the Weight Window which is automatically-generated by Weight Window Generator. To do effective Monte Carlo calculation for the gamma-ray, some energy groups are cut off and weight window is generated by a combination of density reduction method and Weight Window Generator[4]. Track length estimator (TLE) and point detector (PD) are used in tallying process for the particle.

2. MCNP calculation conditions in NFT-14 type cask

2.1. Modeling conditions

NFT-14 type cask is a wet type transportation cask which load 14 PWR spent fuels, and use lead and resin as shielding material. Outline of NFT-14P type cask is shown in **Figure 1**. About this cask, we perform the radiation shielding calculation by using MCNP code with modeling conditions shown in **Table 1**. The calculation model is shown in **Figure 2**.

2.2. Source intensity

2.2.1 Fuels specification

Specifications of spent fuels such as average burn-up (about 33 - 46 GWd/tU) and cooling time (about 840 - 1530 days) are used in source intensity calculation.

Further, reactor operating cycle is taken into account in ORIGEN2.2-upj code calculation. And, a axial burn-up distribution is in common with each fuels.

*Corresponding author. Email: matsumoto@ocl-corp.co.jp

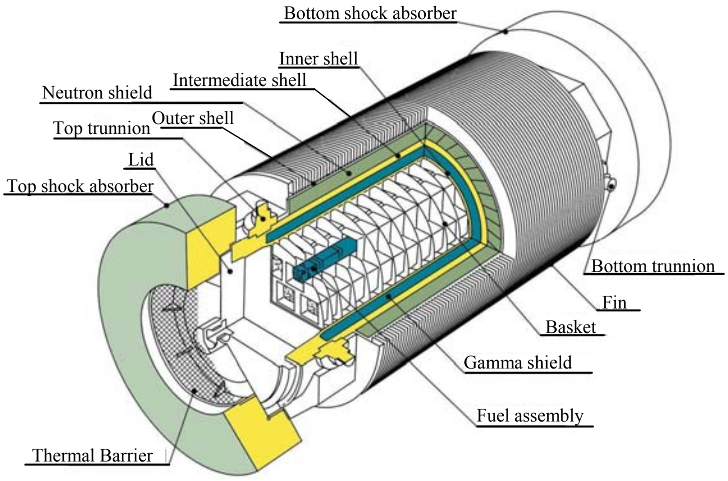

Figure 1. Outline of NFT-14P type cask.

Table 1. Modeling conditions of NFT-14P type cask.

Items	modeling conditions
Fuel assembly	-Fuel assembly is homogenized one by one. -Fuel assembly is divided into top nozzle, top plenum, fuel effective part and bottom nozzle. -Fuel assemblies are positioned closer to bottom side in axial direction, closer to lower side in vertical direction, and in the middle in lateral direction.
Basket	-Basket is basically modeled as real shape.
Cask body	-Cask body is basically modeled as real shape, but valves are ignored. -Expansion of water is ignored for inner water level .
Lid	-Lid is basically modeled as real shape, but lid bolts are simplified.
Shock absorber	-Cover plate is ignored. -Inner ribs are ignored(replaced as wood).
Transport Skid	-Transport Skid is ignored.
Ground	-Ground is ignored.
Composition & Density	-Steels and lead densities are smallest value. -Resin compositions are specific values, and density is 99% of nominal value. (considering initial shrinkage) -Water temperature is taken into account for water density.
Dimensional tolerance	-Dimensions are nominal values.

2.2.2 Results of source intensity calculation

Results or source intensity calculation by ORIGEN2.2-upj code is shown in **Table 2**. Further, total neutron source intensity takes into account the multiplication effect. The effective multiplication factor (k_{eff}) is 0.66.

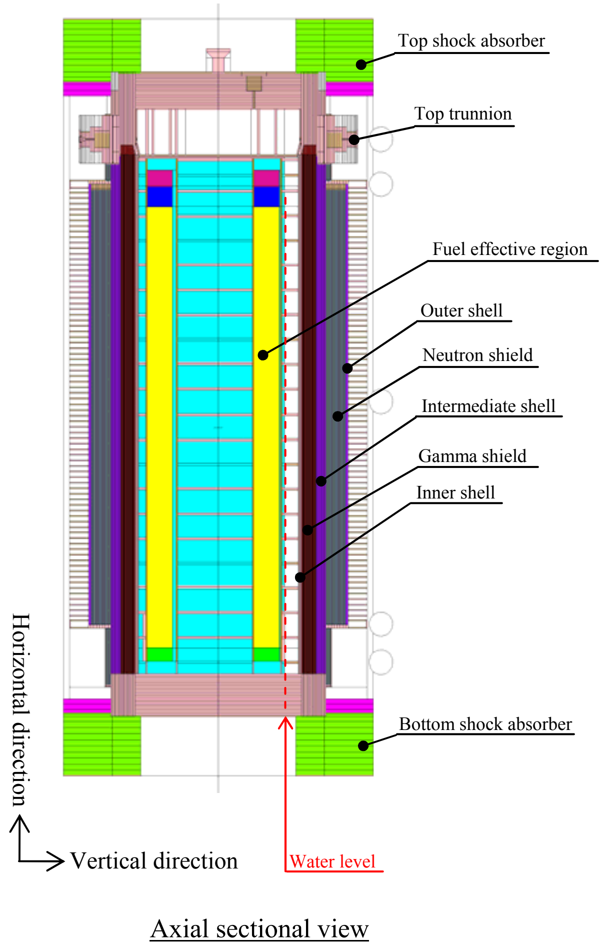

Figure 2. Calculation model of NFT-14P type cask.

Neutron spectrum is Pu^{239} fissile spectrum, which is shown by Watt formula embedded in MCNP code.

For the gamma rays calculation by MCNP code, 1 - 8 groups and 16 - 18 groups of ORIGEN2.2-upj output (18 groups) are cut off, because they don't contribute to the dose rate.

Further, activation gamma source in the end structural materials of the fuel assembly is not evaluated.

Table 2. Source intensity.

Basket Lattice No.	Neutron source (n/s)	Gamma source (photons/s)
#1	2.165E+08	3.436E+15
#2	5.394E+08	5.779E+15
#3	5.394E+08	5.779E+15
#4	8.926E+08	6.952E+15
#5	6.223E+08	7.110E+15
#6	7.745E+08	6.874E+15
#7	8.909E+08	6.447E+15
#8	6.656E+08	6.238E+15
#9	1.030E+09	6.217E+15
#10	1.030E+09	6.217E+15
#11	6.656E+08	6.238E+15
#12	3.553E+08	5.786E+15
#13	7.046E+08	6.630E+15
#14	3.553E+08	5.786E+15

2.3. Evaluation Points

Evaluation points of neutron and gamma ray are shown in **Figure 3** and **Figure 4**.

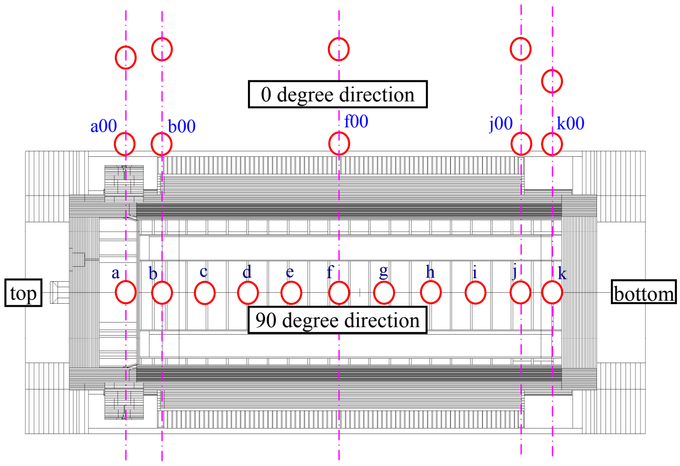

Figure 3. Evaluation points of neutron

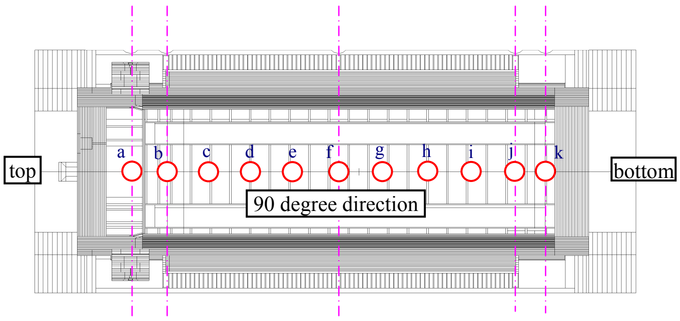

Figure 4. Evaluation points of gamma ray.

2.4. Dose equivalent rate conversion coefficient

For the dose equivalent rate conversion coefficient, conversion coefficient from Air-Kerma to ambient dose (H*(10)) stated in ICRP Publication 74 is used.

2.5. MCNP calculation conditions

MCNP calculation conditions are shown in **Table 3**. At every evaluation points, TLE and PD are used in tallying process for the particle.

Table 3. MCNP calculation conditions.

Items	Content
MCNP Version	Ver. 5
Neutron Library	fsxlb331 (based on JENDL-3.3)
Photon Library	MCPLIB02 (based on ENDF/B-4, EPDL89)
Tally	Point Detector (PD):point type next event estimator Track Length Estimator (TLE):a sphere with radius 10 cm
Variance reduction	Weight Window Method Density reduction method (Three WW calculation steps of 0.6,0.8,1.0 in density coefficient)

3. MCNP calculation results

3.1. Neutron

The comparisons between calculation results of neutron and measurement values are shown in **Figure 5** - **Figure 8**.

At surface in 0 degree direction, calculation value is much greater than measurement value, but the axial distribution profiles are in good agreement

At 1m from surface in 0 degree direction, except for bottom trunnion region, calculation value is greater than measurement value. For bottom trunnion region, calculation value is much greater than measurement value. It is considered that the cause is the neutron source strength of the end portions which is set in a stepwise burn-up distribution.

At surface in 90 degree direction, calculation value is greater than measurement value, and the margin is varied in each axial position and large in top region.

At 1m from surface in 90 degree direction, calculation value and measurement value are almost agreed. And calculation value is a little lower than measurement value in middle and bottom region. There is a wall located near the cask in 90 degree direction when the dose rate was measured, so the large measurement is probably caused by wall reflection of neutrons from bottom trunnion region.

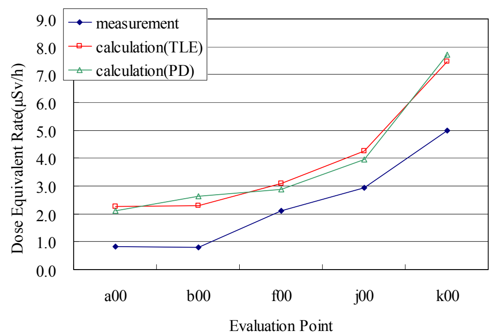

Figure 5. The comparison of neutron dose rate (At surface in 0 degree direction).

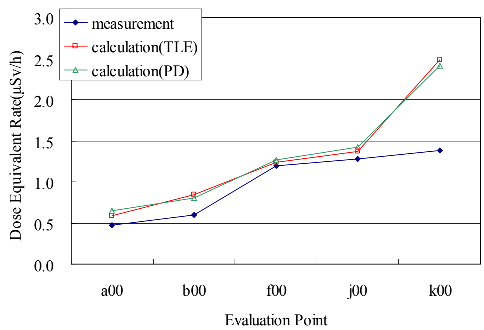

Figure 6. The comparison of neutron dose rate (At 1m from surface in 0 degree direction).

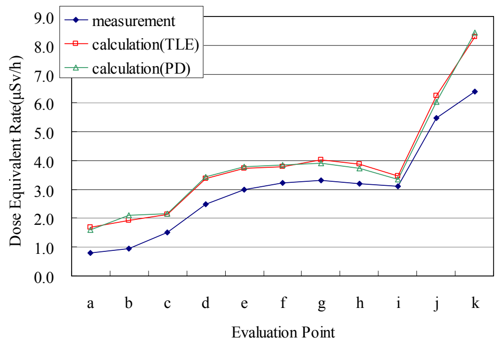

Figure 7. The comparison of neutron dose rate (At surface in 90 degree direction).

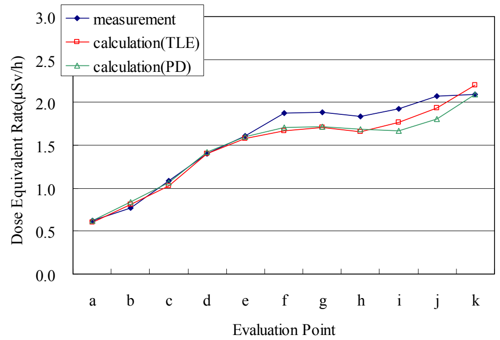

Figure 8. The comparison of neutron dose rate (At 1m from surface in 90 degree direction).

3.2. Gamma rays

The comparisons between calculation results of gamma ray and measurement values are shown in **Figure 9** and **Figure 10**.

At surface in 90 degree direction, calculation value is about 20 percent greater than measurement value in the middle region. Calculation value is lower than measurement value in top region, because the calculation value doesn't include the activation gamma ray from the end structural materials of the fuel assembly.

At 1m from surface in 90 degree direction, calculation value is about 50 percent greater than measurement

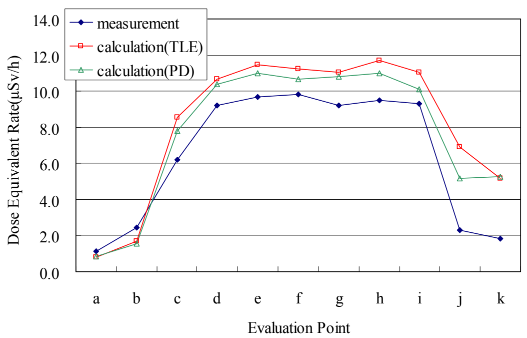

Figure 9. The comparison of gamma ray dose rate (At surface in 90 degree direction).

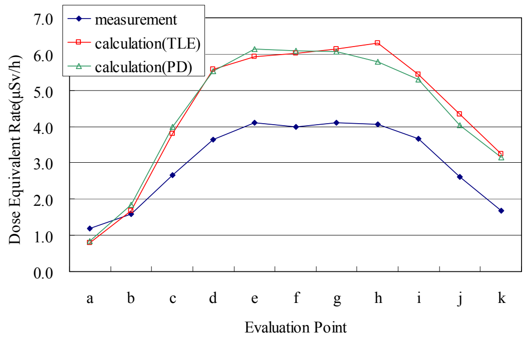

Figure 10. The comparison of gamma ray dose rate (At 1m from surface in 90 degree direction).

value in the middle region. Calculation value is lower than measurement value in top region, because of the same reason as described previously.

4. Conclusion

All calculations pass the relative deviation criteria. With regard to 10 statistical checks provided by the MCNP output, some articles are not satisfied with the criteria in this study. However, the calculated dose rate obtained by the PD and the TLE are in good agreement. In addition, in comparison with the measurements, the axial distribution profiles of the dose rate are in good agreement. These trends represent that the calculated dose rate are acceptable as a reasonable analysis result, even if all statistical checks are not satisfied. In many cases, the calculated dose rate is larger than the measured dose rate, and therefore the calculation procedure presented in the Guideline is conservative and useful for the purpose of the safety analysis.

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