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Dose Equivalent Rate Benchmark Calculations of a Dry Storage Cask for Spent Fuel by 3D Monte Carlo code

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ABSTRACT

MHI has been trying to apply a new dry cask design method with an advanced 3D shielding calculation method instead of conventional 2D calculation methods which have been used in licensing examination by a regulatory agency. The 3D calculation method is to utilize MCNP Monte Carlo calculation code that can evaluate dose equivalent rate of cask with complex structure more sophisticatedly and accurately than 2D calculation codes. Although it has been used in detailed calculation for reference, it can be practical to use the 3D Monte Carlo calculations because of today's increasing computation power and capacity. To apply 3D Monte Carlo calculations to the licensing of dry casks, validation procedure has to be performed. However, only a few "detailed experimental data" are available to validate 3D Monte Carlo codes for dry cask application. Therefore, MHI conducted an experiment with a trial small dry cask which contains a spent fuel and shielding structures similar to actual dry casks in order to measure dose equivalent rates. MHI then performed 3D Monte Carlo calculations by MCNP code with a detailed analysis model including spent fuel assembly and cask structure shielding with chemical compositions of materials. The calculated dose equivalent rates agreed well with the measured values within 20% deviations. It is validated that the MCNP calculation is applicable to dose equivalent rate calculations for dry cask licensing.

INTRODUCTION

In Japan, 9 PWR plants have restarted in recent three years, and the amount of spent fuels has increased accordingly. The main storage method of storing spent fuels in Japan is to store in a spent fuel pool. Consequently storing spent fuel assemblies will be a critical issue to be solved in the near future because spent fuel pools in each plant will be filled up with their spent fuel assemblies generated by plant operation. This problem must be resolved for stable operation of plants and nuclear fuel cycle. One candidate solution for storing spent fuel assemblies is to store them in dry casks instead of spent fuel pools, because dry casks provide high sealing integrity of radioactive material and structural toughness. In Japan, spent fuel storage in dry casks is attracting attention and is being introduced.

In shielding calculation of dry casks, 2D calculation codes (e.g. DOT3.5[1] and DORT[2]) have been used to calculate dose equivalent rates for safety analysis report. These codes restrict calculation models to cylinder, and spent fuel assemblies and basket areas within casks are homogenized. Because complicated structures such as trunnions and valves cannot be treated as a single analytical model, it is necessary to create a separate model for that part, and to calculate by connecting flux around trunnions and valves to the model. When trunnions or valves are to be modeled by 2D calculation code, conservative models have to be used because of the constraint of geometrical modeling as a cylinder. For trunnions and valves, it is necessary to estimate the effects of radiation streaming, independently.

On the other hand 3D Monte Carlo calculation codes with precise modeling of actual shapes can handle complex structures with a single analytical model. Thus it can be expected to calculate dose equivalent rates more realistic than 2D calculation codes. It is necessary to performed validation procedure in order to apply the 3D calculation codes as MCNP[3] to shielding calculation of dry casks. An experiment was conducted with a trial small dry cask, which contains a spent fuel and shielding structures similar to actual dry casks in order to measure dose equivalent rates at its surfaces. Then, the conditions required for the benchmark analysis such as fuel specifications, cask dimensions and surrounding structures, were modeled in detail, and the results were compared with the measured values to confirm the validity.

FUEL LOADED DRY CASK EXPERIMENT

The trial small dry cask used in the present measurement experiment has a height of about 5.2 m and a diameter of about 1.7 m to accommodate 2 PWR spent fuel assemblies. The structure of this cask is similar to an actual dry cask consisting of resin, carbon steel and stainless steel.

In the present experiment, 1 unit of 17 x 17 type spent fuel assembly with burnup of about 43 GWd/t and cooling period of about 24 years was loaded in the cask.

The measurements of neutron and gamma dose equivalent rate were performed on the cask surface. A rem counter (Made by Aloka, model number: TPS-451) and an ionization chamber type survey meter (Made by Aloka, model number: ICS-301) were used for neutron and gamma ray measurements, respectively. The average of the three measurements was employed for each point. Figure 1 shows the drawings of horizontal and vertical cross section s of the test cask including measurement points.

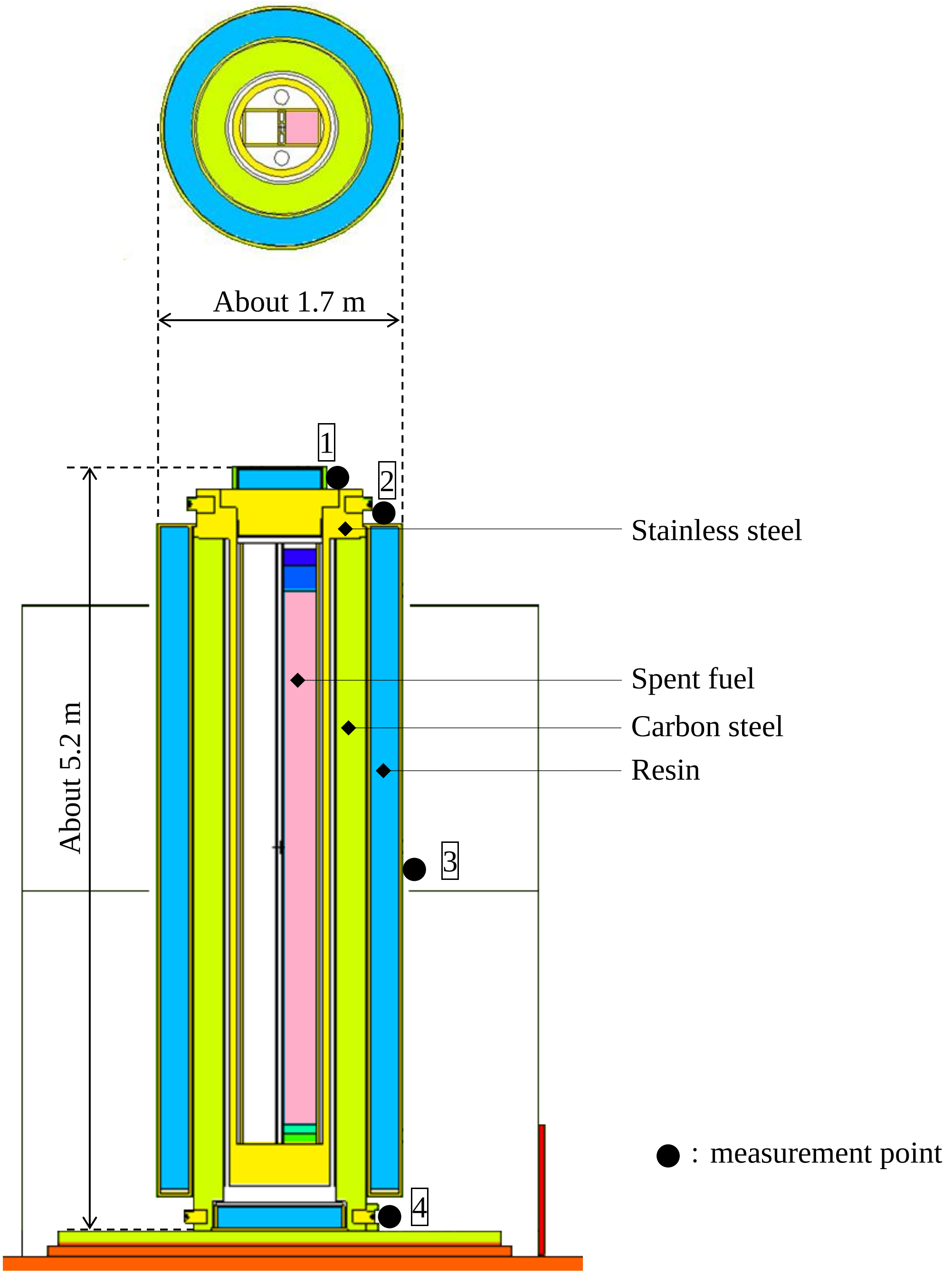

Figure 1. Measurement environment and measurement point

CALCULATION METHOD

The source strengths of the active fuel were predicted by ORIGEN2.2UPJ[4] code based on the spent fuel specifications for neutron and gamma-ray, respectively. The library used in the calculation was PWR41J33. The burnup distribution in the axial direction was considered with the actual burnup data.

The gamma-rays from the irradiated fuel assembly structure were induced by ^{60}Co. Each gamma-ray was estimated using the initial ^{59}Co content in the fuel structure, the irradiation time and the cooling time as the following Eq.(1).

$$\mathrm{A} = N_0 \cdot \sigma \cdot \phi \cdot \{1 - exp(-\lambda T_1)\} \cdot exp(-\lambda T_2) \qquad (1)$$

where

A : Radioactivity (^{60}Co) [Bq],
N_0 : Number of target atoms (^{59}Co) [atoms],
σ : (n,γ) reaction cross section ^{59}Co with 2200 m/s neutrons [cm^2],
ϕ : Reactor thermal neutron flux [neutrons/(cm^2 s)],
λ : Decay constant [/day],
T_1 : Irradiation days [day],
T_2 : Cooling days [day].

Table 1 shows the calculation results of source strength.

Table 1. Source strength calculation results

Position		Source
Total neutron source strength* (Consider effective multiplication factor)		1.2E+08 neutrons /s
Total fuel gamma-ray source strength*		2.8E+15 photons /s
Fuel assembly structure gamma-ray source strength	upper nozzle	3.5E+10 photons /s
	upper plenum	2.7E+11 photons /s
	active fuel	1.3E+10 photons /s
	lower end plug	2.8E+10 photons /s
	lower nozzle	1.3E+11 photons /s

*It is the source strength of the whole fuel. In the calculation by MCNP, the source strength was set according to the burnup distribution in the axial direction.

Using the calculated source strength, dose equivalent rates around the cask was evaluated by MCNP. Table 2 shows the MCNP calculation conditions. The cask in this calculation was modeled almost the actual shape, and the surrounding structure of experimental room such as floor and walls was precisely modeled to consider effects of radiation scattering.

Table 2. MCNP calculation condition

<table>
<tr><th colspan="3"></th><th>Condition</th></tr>
<tr><td colspan="3">Code version</td><td>MCNP5 ver.1.60</td></tr>
<tr><td colspan="3">Calculation model</td><td>Based on Figure 2.</td></tr>
<tr><td colspan="3">Cross section library</td><td>FSXLIB-J33 [5]
MCPLIB84</td></tr>
<tr><td rowspan="3">Energy spectrum</td><td colspan="2">neutron</td><td>Watt type ^{244}Cm spontaneous fission neutron spectrum</td></tr>
<tr><td rowspan="2">gamma ray</td><td>active fuel</td><td>Use results from
18 groups structure of ORIGEN2</td></tr>
<tr><td>activation of fuel assembly structure</td><td>1.17MeV : 50%
1.33MeV : 50%</td></tr>
<tr><td colspan="3">Effective multiplication factor</td><td>Separately evaluated by critical analysis using KENO-VI [6]</td></tr>
<tr><td colspan="3">Dose equivalent rate conversion factor</td><td>1 cm dose equivalent rate conversion factor based on ICRP Pub. 74</td></tr>
<tr><td colspan="3">Variance reduction method</td><td>Weight window</td></tr>
<tr><td colspan="3">Tallies</td><td>F4 tally (track length estimator)
neutron : sphere 20 cm in diameter
(equivalent to rem counter)
gamma-ray : sphere 10 cm in diameter
(equivalent to an ionization chamber)</td></tr>
</table>

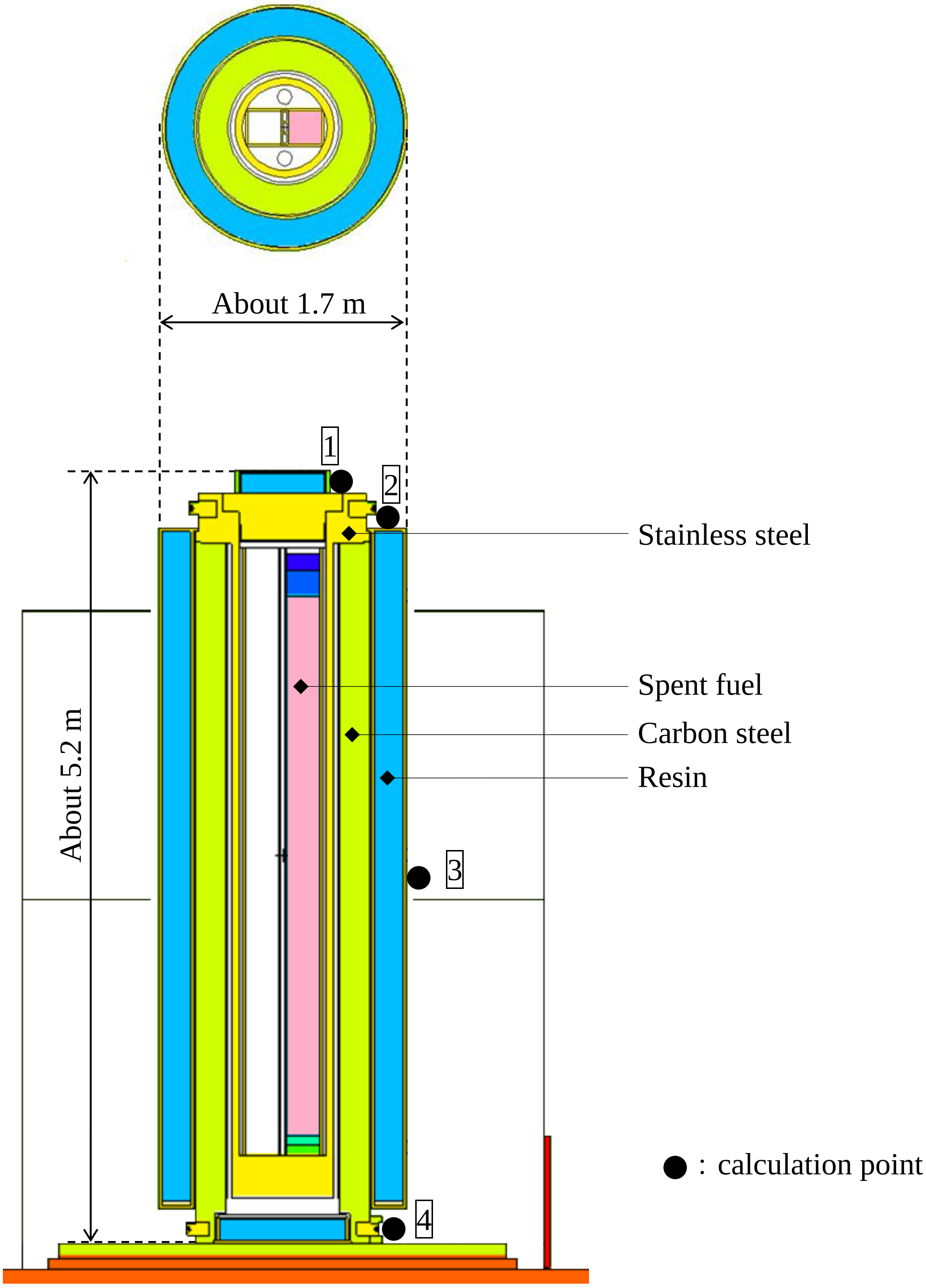

Figure 2. MCNP calculation model for the experiment

RESULTS and DISCUSSION

Tables 3 and 4 show the comparison between experimental and calculated values of dose equivalent rate. It is found that for neutron dose equivalent rate, the calculated / experimental (C/E) ratios are within 20%. The reason why the dose equivalent rate at measurement point 3 is lower than other measurement points is that there is a neutron shielding refereed as “Resin” in Fig.2 between the spent fuel and the measurement point.

For gamma-ray dose equivalent rate, the experimental values are below the detection limit in areas except the center of fuel assembly. There are two causes expected; one is because the effective shielding thickness, which is to defined here as length between source and measurement point, of the other areas are relatively thicker than the center area. The other is because the gamma ray source strength is low due to low burnup near fuel top and bottom regions. The C/E ratio at the center of the cask is within 20%.

These results show that dose equivalent rate of dry casks by MCNP was successfully calculated.

Table 3. Experimental and calculated results (neutron dose equivalent rate)

Measurement point	Experimental value (μSv/h)	Calculated value (μSv/h)	Calculated / Experimental
1	9.93±0.17	10.20±1.8E-02	1.03±0.02
2	9.71±0.13	9.01±3.2E-02	0.93±0.01
3	0.47±0.02	0.52±9.8E-03	1.11±0.04
4	9.66±0.18	11.33±3.7E-02	1.17±0.02

Table 4. Experimental and calculated results (gamma-ray dose equivalent rate)

Measurement point	Experimental value (μSv/h)	Calculated value (μSv/h)	Calculated / Experimental
1	N/D*	-	-
2	N/D*	-	-
3	2.5±0	2.25±0.04	0.90±0.02
4	N/D*	-	-

*below the detection limit 0.5 μSv/h

CONCLUSION

In order to apply MCNP calculations to cask design, a experiment was performed to obtain the benchmark result for the MCNP validation. In the experiment, a trial small dry cask similar to an actual dry cask was used. The dose equivalent rates around the trial small dry cask with an actual spent fuel were measured in this experiment.

The dose equivalent rate was calculated by MCNP with detail modeling. The C/E ratios of both neutrons and gamma-rays were within 20%. It is validated that the MCNP calculation is applicable to dose equivalent rate calculations for dry cask licensing.

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