EXACT COMPARISON OF DOSE RATE MEASUREMENTS AND CALCULATION OF TN12/2 PACKAGES

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SUMMARY

Both of dose rate measurements of TN 12/2 package and calculations by Monte Carlo code MORSE in SCALE code system and MCNP were performed to evaluate the difference between the measurement and the calculation and finding out the cause of the difference. The calculated gamma-ray dose rates agreed well with measured ones, but calculated neutron dose rates overestimated more than a factor of 1.7. When considering the cause of the difference and applying the modification into the neutron calculation, the calculated neutron dose rates become to agree well, and the factor decreased to around 1.3.

INTRODUCTION

The NEACRP organized an inter-comparison of codes for the calculation of neutron and gamma-ray dose rates for packages (A.F.Avery and H. Locke 1994). Twenty codes including Monte Carlo, Discrete Ordinates, and Point Kernel were applied in this inter-comparison, but there was a large variation during these calculations and it was concluded that there was a difficulty of the absence of any "correct" answer for the theoretical benchmarks or even in the experimental benchmark, because of the lack of precise data for calculation such as actual dimensions and densities of a package used in the measurements.

Considering the availability of information of a package, spent fuels and measurements, the authors performed a dose rate measurement of TN 12/2 package and then made calculations using data such as the detailed source specification, the details of the structure of the package, and the measurement conditions. The measured results were compared with the calculated figures to clarify the cause of discrepancies between the measurement and the calculation.

MEASUREMENTS

Configuration of the TN 12/2 Package

The TN 12/2 package is a transport package to convey 12 PWR spent fuels from a Japanese reactor site to COGEMA in France (G. Sert 1983). The cross-sectional view of the TN 12/2 package is shown in Fig.1. This is one of the largest transport packages in the world. There are shock absorbers at the top and bottom on either side of the finned area, each consisting of a stainless steel shell packed with balsa wood. The main body consists of forged carbon steel, surrounded by a resin layer, which is a neutron shield. The resin layer is penetrated by copper cooling fins to transfer the decay heat to the outer surface of the package. The basket within the package is composed of cast aluminum alloy to transfer the heat from spent fuels to the inner surface of the body.

Detectors and Measured Point

Gamma-ray dose rates were measured by an ionization chamber and neutron dose rates were measured by a rem-counter. The locations of the measured points are shown in Fig.2. According to the preliminary calculations, these were the points where higher dose rates were expected. The background dose rates were measured after the packages were moved outside. The values obtained were less than 0.1 mR/h for the gamma-ray dose rate and negligibly small for the neutron dose rate.

Specifications of the Fuels

Measurements were performed for two TN 12/2 packages. Before loading the spent fuels into the packages, the specifications of the spent fuels to be loaded into these two packages were investigated and the location of each spent fuel was arranged to get a similar dose rate distribution for both packages and to have a uniform distribution in each package. A summary of the specifications for the spent fuel assemblies loaded into these two TN 12/2 packages is shown in Table 1. As the enrichment and burnup of the "B-" and "C-" types of fuel assemblies were very similar and had the same operation pattern, these fuel assemblies had almost the same characteristics with respect to radiation source. Therefore, "B-" and "C-" types of fuel assemblies could be treated as in the same category. Then, eight of the type "A" fuel assemblies were loaded separately into the four inner side lodgements of each package, and the other types "B" and "C" fuel assemblies were loaded into the eight outer side lodgements of each package.

Results of Measurement

The measured dose rates are summarized in Tables 2 and 3. The highest dose rate appeared toward the bottom direction, but almost the same with the dose rates on the sides. The dose rate around the lid was about one order of magnitude lower than that in the bottom area. The axial gamma-ray and neutron dose rate distributions along the sides of the package are shown in Figs. 3 and 4, respectively. As intended before loading the spent fuels into the packages, the measured dose rates of these two packages were almost the same.

CALCULATIONS

As the spent fuel assemblies loaded in the two packages have the same fuel loading pattern and the results for the dose rate measurements were almost the same, the analysis was based on the fuel specifications for only the one of two packages.

Source Intensity

The source intensity of each spent fuel assembly was calculated using the ORIGEN2 code (A. G. Croff 1980). The reactor library used in the ORIGEN2 code was the PWRUS (S. B. Ludwig and J. P. Renier 1989). The irradiation history of each of the "A-", "B-" and" C-" type fuel assemblies comes from cycle patterns shown in Table 1.

Calculation Conditions

The SAS4 driver in SCALE code system (SCALE 1993) was modified to get the dose rate distribution of a whole package surface (H. Taniuchi 1995). The SAS4 driver uses the Monte Carlo code MORSE for shielding calculation. In addition, the MCNP (Briesmeister, J. F,Ed. 1993) was used to check the difference in results between the two Monte Carlo codes. The cross-section library for SAS4 was the coupled 27-neutron-group and 18-gamma-ray-group library that is provided in the SCALE system, and the MCNP used the ENDT5T2 library for neutron and the MCPLIB2 library for gamma-ray. The flux-to-dose-rate conversion factors were from ANSI/ANS 6.1.1-1977. For the neutron source spectrum, spontaneous fission spectra of ^{242}Cm and ^{244}Cm, and (α,n) neutron spectra of ^{242}Cm, ^{244}Cm and ^{238}Pu were considered. A typical axial burn-up profile for the spent fuel assembly provided in SAS4 was assumed in both the neutron and the gamma-ray calculations. The effective multiplication factor for this configuration was assumed to be 0.15 from the reference (H. F. Locke 1992).

For the gamma-ray source from fission products, the eight energy groups source spectra derived from the ORIGEN2 code were converted to the energy structure of the 27-neutron-group SCALE library. For the activated ^{60}Co gamma-ray source in the end pieces of fuel assemblies, 1.17 and 1.33 MeV gamma ray energies were considered.

For the calculation of the radial direction of TN 12/2 package, the actual neutron shield region has a complicated geometry and the thickness of the region varies according to the angle, as shown in Fig.1. To simplify the calculation model, the minimum thickness of the neutron shield (resin) was used in the calculation.

COMPARISON AND DISCUSSION

Comparison of Results

The fractional standard deviation of SAS4 and MCNP calculation was less than 15%, mostly less than 5% for neutron calculation. The comparison of measured and calculated gamma-ray dose rates is shown in Table 2 and the comparison along the z-axis is shown in Fig.3. For the figures for the sides of the package, the agreement of the measured results with the results calculated by SAS4 and MCNP was generally good. There were some discrepancies in the figures for both end surfaces. In these parts, a dose contribution of ^{60}Co was large according to the calculation.

The comparison of measured and calculated neutron dose rates is shown in Table 3 and the comparison along the z-axis is shown in Fig.4. With respect to neutron dose rate, the SAS4 and MCNP had a little different results. On the sides, the SAS4 calculation overestimated by a factor of two at the surface and 1.7 at 1 m from the surface, the MCNP also overestimated but the factor is less. At the top, the overestimation was much larger on the surface. At the bottom, there was a good agreement at the surface but underestimation at 1 m from the package surface.

Discussion of the calculation conditions

As described in the above section, there were some discrepancies between the measured and calculated dose rates, especially for neutron. To clarify the reason for the discrepancies, the following investigations were performed.

1.Thickness of resin and density of balsa wood
The actual thickness and density of each shield were checked to find out the differences in resin thickness and in the density of balsa wood used in the calculations and the actual ones. The average resin thickness was actually 1.2cm thicker than the minimum thickness used in the calculation. With respect to the density of the balsa wood, the minimum density of 0.20 g/cm^3 was used in the calculation, but the actual density of balsa wood of these packages was 0.21 g/cm^3. After changing the figures for thickness of the neutron shield and the density of balsa wood, revised calculations were performed using SAS4 for neutron dose rate. The calculated neutron dose rate dropped about 30% for the radial direction and about 10% towards the bottom.

2. Effect of burnup profile
The calculation using a flat burn-up distribution showed about 0.7 times the neutron dose rate using the typical burn-up distribution for the radial direction and about 1.7 times that for the bottom region. The gamma-ray dose rate was not so sensitive with this distribution for the radial direction, but for the bottom region, the flat burn-up distribution gave dose rate a factor of 4 higher. The axial neutron source intensity varies greatly depending on the burnup profile, and the top and bottom gamma-ray dose rates are closely correlated with the burnup profile.

3. Source intensity of ^{60}Co
The gamma-ray source intensity at the endfittings were calculated on the basis of the maximum impurity level of ^{59}Co, but the actual impurity level might be a factor of 2 or 3 less than the maximum level. Unfortunately, there was no available data for the spent fuel assemblies loaded in these TN 12/2 packages. The evaluation of ^{60}Co intensity is very important when storing spent fuel assemblies with much longer cooling times such as more than 5 years.

4. Difference in detector response
The response of the neutron rem-counter does not demonstrate the same energy dependence as the ANSI/ANS 6.1.1-1977 flux-to-dose conversion factor used in the calculation. The effect of this difference was checked by a MCNP calculation, with the results showing an effect of less than 5%. An angular dependence of the response for neutron measurements shows a much greater effect on the surface dose rate. According to the Reference (H. F. Locke 1992), the surface neutron dose rates turn out to be about 0.9 times the calculated ones for the angular dependence of the response.

5.Surface dose rate
In the measurements, the rem-counter can not measure the exact surface dose rate because the size of the rem-counter is usually large and the dose rate decreases rapidly near the surface. The dose rates at the effective center of the rem-counter were calculated by SAS4 and the results obtained showed a dose rate 10% lower than those at the surface for neutron.

When evaluations take into account (1) the actual average thickness of the resin and the actual density of the balsa wood, (2) the effect of the angular dependence of the response on the neutron measurements for the surface dose rate, and (3) the effective center of the detector, the agreement between the measured and calculated neutron dose rates improves to within a range of 30%. Furthermore, if the actual burnup profile of the spent fuels and the actual impurity level of ^{59}Co were available, a much closer agreement could be expected.

CONCLUSION

The neutron and gamma-ray dose rate measurements of two TN 12/2 packages containing short-cooled spent fuels were performed. Subsequently, calculations were made using the SAS4. The causes of the differences between measured and calculated results obtained were discussed and estimated. The agreement between the measured and calculated neutron dose rates improves to within a range of 30% when considering the cause of the difference and applying the modification into the neutron calculation.

ACKNOWLEDGMENTS

The authors wish to express their thank to the Kansai Electric Power, Co. Inc. for allowing us to use their data for this study. The authors also appreciate the assistance of Mr. M. Mimura, Mr. T. Tsuboi and Ms. I Imanishi of CRC Research Institute, Inc of preparing the data for calculation and their contribution to complete the calculations.

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Table 1 Specifications for spent fuel assemblies loaded into TN-12/2 packages

Assembly type	Initial enrichment (%)	Cycle No.	Cumulative burnup (MWd/tU)	Specific power (MW/tU)	Irradiation time (day)	Cooling time (day)
A	2.69	1	14316	22.0	650	212
		2	23588	30.1	308	723
		4	33204	29.9	322	546
B	3.25	2	7544	24.5	308	117
		3	16771	35.1	263	291
		4	28077	35.1	322	546
C	3.24	2	9599	31.2	308	117
		3	19174	36.4	263	291
		4	29371	31.7	322	546

Table 2 Comparison of masured and calculated gamma-ray dose rates around TN-12/2 package

Measured point No.[a]	Gamma-ray dose rate (mR/h)[d]							
	Surface				1 m from Surface			
	Meas.-1	Meas.-2	SAS4M	MCNP	Meas.-1	Meas.-2	SAS4M	MCNP
Side								
1	1.9	1.8	2.61[1.41][c]	3.61[1.95]	1.5	1.8	2.06[1.25]	2.53[1.53]
2	5.9	5.7	6.31[1.09]	8.03[1.38]	2.5	2.5	2.43[0.97]	3.50[1.40]
3	6.7	6.5	7.87[1.19]	8.43[1.28]	3.0	3.1	2.85[0.94]	3.90[1.28]
4	6.8	6.7	7.08[1.05]	8.51[1.26]	3.3	3.4	3.53[1.06]	4.04[1.21]
5	6.7	6.9	8.07[1.19]	9.92[1.46]	3.3	3.3	3.36[1.02]	4.50[1.36]
6	6.7	6.6	9.85[1.48]	9.65[1.45]	3.2	3.3	3.04[0.93]	4.20[1.29]
7	6.7	6.3	8.19[1.26]	7.54[1.16]	2.9	2.9	2.92[1.01]	3.44[1.19]
8	5.2	4.8	5.34[1.07]	7.23[1.45]	2.1	2.4	2.41[1.07]	2.71[1.20]
9	0.9	0.9	2.41[2.67]	1.10[1.22]	1.0	1.2	2.11[1.92]	1.47[1.34]
Top								
16	0.7	0.7	1.84[2.63]	1.76[2.16]	0.3	0.3	0.81[2.70]	0.44[1.47]
Bottom								
18	- (7.4)[b]	- (7.0)	8.31 -	7.31 -	- (2.80)	- (2.80)	3.07 -	2.49 -
19	6.7(6.3)	6.4(5.8)	9.23[1.41]	7.07[1.08]	3.8(3.30)	3.7(3.10)	3.58[0.95]	2.66[0.71]
20	8.7(7.4)	8.5(7.5)	8.29[0.96]	6.91[0.80]	- (3.20)	- (2.90)	3.07 -	2.49 -

[a] These numbers correspond to the numbers in Fig. 2

[b] Values in parentheses are measured dose rates obtained a month later than the original measurement date.

[c] Values in brackets are the C/E ratio against the average of two measured dose rates.

[d] 1 mR/h = 10 μ Sv/h

Table 3 Comparison of measured and calculated neutron dose rates around TN-12/2 package

Measured point No.[a]	Neutron dose rate (mrem/h)[d]							
	Surface				1 m from Surface			
	Meas.-1	Meas.-2	SAS4M	MCNP	Meas.-1	Meas.-2	SAS4M	MCNP
Side								
1	1.8	1.6	2.59[1.52][c]	1.91[1.12]	0.8	0.9	1.41[1.65]	0.70[0.82]
2	1.7	1.7	3.20[1.88]	2.01[1.18]	0.8	1.0	1.59[1.76]	1.05[1.17]
3	2.0	1.8	3.96[2.08]	2.88[1.52]	0.9	0.9	1.66[1.84]	1.08[1.20]
4	2.0	2.2	4.45[2.12]	3.13[1.49]	1.2	1.1	1.81[1.57]	1.25[1.09]
5	2.2	2.2	4.56[2.07]	3.43[1.56]	1.1	1.0	1.75[1.67]	1.12[1.07]
6	2.1	2.0	4.33[2.11]	3.52[1.72]	0.9	1.0	1.66[1.75]	1.25[1.32]
7	1.7	1.6	3.78[2.29]	2.83[1.72]	0.8	1.0	1.43[1.59]	0.98[1.09]
8	1.3	1.3	2.23[1.72]	1.79[1.38]	0.7	0.8	1.10[1.46]	0.84[1.12]
9	1.1	1.3	1.63[1.36]	1.46[1.22]	0.5	0.6	0.82[1.50]	0.60[1.09]
Top								
16	0.3	0.4	1.42[4.06]	0.51[1.46]	0.2	0.3	0.34[1.36]	0.15[0.60]
Bottom								
18	- (4.0)[b]	- (3.5)	4.66[1.24]	3.77[1.01]	- (1.20)	- (1.20)	1.14[0.95]	0.98[0.82]
19	3.5(4.0)	3.6(3.9)	6.08[1.62]	4.22[1.13]	1.2(1.30)	1.1(1.30)	1.30[1.06]	0.98[0.80]
20	4.2(4.0)	4.0(3.9)	4.67[1.16]	3.80[0.94]	- (1.30)	- (1.20)	1.13[0.91]	0.96[0.77]

[a] These numbers correspond to the numbers in Fig. 2.

[b] Values in parentheses are measured dose rates obtained a month later than the original measurement date.

[c] Values in brackets are the C/E ratio against the average of two measured dose rates.

[d] 1 mrem/h = 10 μ Sv/h

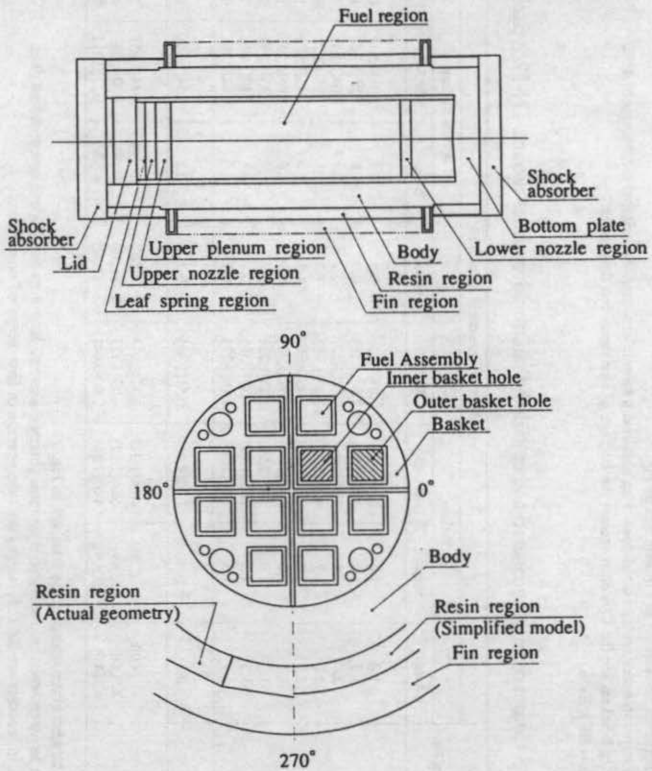

Fig. 1. Cross-sectional View of TN 12/2 package

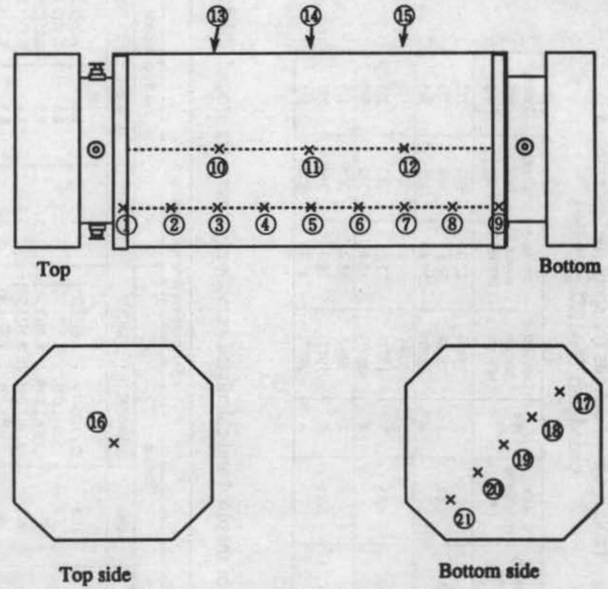

Fig. 2. Dose Rate Measurement Position on TN 12/2 Package

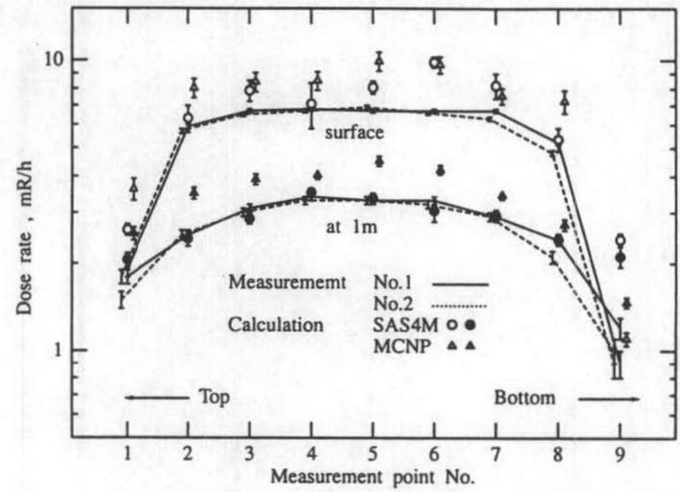

Fig. 3. Comparison of gamma-ray dose rate of a TN 12/2 package (side)

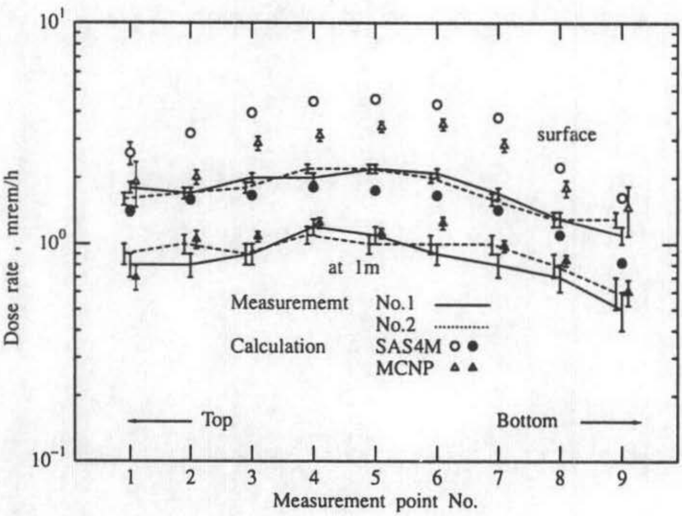

Fig. 4. Comparison of neutron dose rate of a TN 12/2 package (side)

SESSION 11.3

Risk Assessment