EVALUATION BY PHYSICAL MEASUREMENTS OF SHIELDING ANALYSIS METHOD USED FOR TN-12/1

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1. Introduction

TN-12/1 package containing 12 spent fuel assemblies have been shipped from PWR power plants in Japan to overseas reprocessing facility in accordance with the regulations for safe transport of radioactive materials.

The Kansai Electric Power Co., Inc. has introduced a TN-12/1 packaging using a new basket that was designed by the Transnucleaire S.A. in France. In this basket, the conventional sintered B_4C/Cu plates have been replaced by a boron/aluminum alloy (P. Meyer et al., 1989), and stainless steel shielding blocks have been used as some parts of the basket sector.

This replacement was accompanied by a revision of the documents of safety analyses for TN-12/1 package including shielding calculations.

After introducing the new basket, the Company carried out the shipping of the first TN-12/1 packages containing spent fuel assemblies. During the shipping process, the Company jointly with Transnuclear Ltd. measured the dose equivalent rates around the TN-12/1 packages.

The measured values were compared with the values determined by the evaluation method for dose equivalent rate in the shielding analysis. This comparison demonstrated that the calculated values coincide with or slightly exceed the measured values. At the same time, the shielding performance of the new basket was also checked.

2. Package

The general specifications of TN-12/1 package are shown in Table 1 and the transport packaging is illustrated in Figure 1.

Packages are transported by vehicles and special use vessels.

During transportation, each package is placed on a transport frame in the horizontal position and secured on the top and bottom trunnions. In addition, thermal barrier to prevent access is provided over the fin and trunnion regions (Figure 2).

3. Measurement of dose equivalent rate

The dose equivalent rates for gamma-ray and neutron were measured for the package which contained the fuels as shown in Table 2.
Following devices were used for the measurement.

a. Device for measuring gamma-ray dose equivalent rate
   Survey meter: Ionization chamber, ICS-303 (Aloka)
b. Device for measuring neutron dose equivalent rate
   Survey meter: Rem counter, Studvik 2202D.

Dose equivalent rates were measured at the surface and at 1m from the surface of the TN-12/1 package.
It is confirmed that the measured maximum dose rates are enough below the Japanease regulatory limit, 2mSv/h at the surface and 100μSv/h at 1m from the surface of the package (Details of the results of measurement are discussed in para. 5).

4. Calculation of dose equivalent rates

The radiation sources which are considered in the shielding analysis for the TN-12/1 package are gamma-ray and neutron from the spent fuel assemblies contained in the package.

(1) Radiation source

a. Gamma-ray sources

Gamma-ray sources are classified into the two types:

(a) Gamma-ray from fission products
(b) Activated radiation source in the structural material of the ends of fuel assemblies (Mainly ^{60}Co)

The calculations were performed using ORIGEN-2 code.

b. Neutron sources

Neutron sources are classified into the two types:

(a) Neutron from spontaneous fission

(b) Neutron from (α, n) reaction

These intensity of neutron sources was calculated using ORIGEN-2 code. The energy spectrum of neutrons was calculated using Cranberg's equation for ^{235}U. The multiplication factor (Keff) is taken into account for the neutron source intensity.

(2) Calculative specifications

a. Calculation of gamma-ray shielding

QAD code was used for calculating the gamma-ray shielding.

The radiation source region was divided into 12 subregions corresponding to the number of fuel assemblies contained.

b. Calculation of neutron shielding

DOT3.5 code was used for calculating the neutron shielding.

The radiation source region was assumed to be cylindrical. The radius of this cylindrical region was fixed so that fuel elements may be circumscribed to the basket lodgements.

5. Comparison of calculated values with measured values

The dose equivalent rate distribution in the longitudinal direction are shown in Figure 3 and 4. In these figures, the dose equivalent rates are normalized by the maximum value of calculation or measurement.

These figures illustrate:

- There are some discrepancies near the top of the body between calculated values and measured ones. This is due to the conservative assumption for the calculation. (In the calculation, fuel assemblies are close both to the lid and bottom plate, but, actually, they are close only to the bottom plate.)
- However, generally, there is a good agreement between the calculated and measured values.

Then, maximum dose equivalent rates both of the calculation and the measurement at the surface and at 1 meter from the surface of the package are in Table 3.

This table illustrates:

- For the maximum dose equivalent rates for gamma-ray plus neutron, the calculated values are larger than the measured ones. The calculation method is conservative and the validity of the method was confirmed.

6. Shielding performance of the new basket

The stainless steel blocks are incorporated in each sector of the new basket for TN-12/1 packaging for the gamma-ray shielding.

The gamma-ray dose equivalent rates at 1 meter from the surface of the package were flatly distributed in the circumferential direction as shown in Figure 5. And at the surface, similar results are obtained as shown in Figure 6, but, the small peaks of the surface dose equivalent rate were observed in the direction corresponding to the stainless steel blocks.

In order to evaluate the shielding effect of stainless steel blocks, QAD calculations were carried out for the basket with and without stainless steel blocks.

As a result, the peak value of the dose equivalent rates can be reduced by about 40% by incorporating stainless blocks in the new basket.

7. Conclusion

(1) The comparison of the measured values with the calculated values has shown that there is a good agreement between them and the calculation method is conservative.

(2) It is confirmed that the design of the new basket incorporating stainless steel blocks is appropriate from the viewpoint of gamma-ray shielding.

Reference

P.Meyer et al., A New Basket for Transport/Storage Casks, PATRAM'89 (1989)

Table 1 General specification of TN-12/1 package

Name of package	TN-12/1
Type of package	Type-B(M) fissile package
Transport Index	≤ 10
Maximum gross weight of package	Approximately 100 ton
Outer size of packaging (Outer diameter ×Total length)	Approximately 2.5m×5.9m (Including Shock absorbing cover)
Weight of packaging	Approximately 92 ton
Main material of packaging	
Body	Carbon steel, Resin, Copper
Lid	Carbon steel
Basket	Boron/Aluminum alloy Stainless steel
Fuel assembly supporting spacer	Stainless steel
Shock absorbing cover	Wood, Carbon steel

Table 2 Specification of the actual spent fuel assemblies loaded in the TN-12/1 package

	Intial U weight (kg)	Intial ^{235}U enrichment (%)	Burn-up (MWD/MTU)	Cooling time (day)
Range	440 ～ 460	3.2 ～ 3.3	28,000～33,000	500 ～ 600
Average	～ 450	～ 3.3	～ 30,000	～ 550

(Note) There were afew spent fuel assemblies outside of the above range.

Table 3 Results of the dose equivalent rates that were measured and calculated

	Dose equivalent rate (μSv/h)					
	Surface			At 1m from the surface		
	Gamma-ray	Neutron	Total	Gamma-ray	Neutron	Total
Body	21	74	95	20	18	38
	18	35	53	9	15	24

(Note) The upper values were calculated, the lower ones were measured.

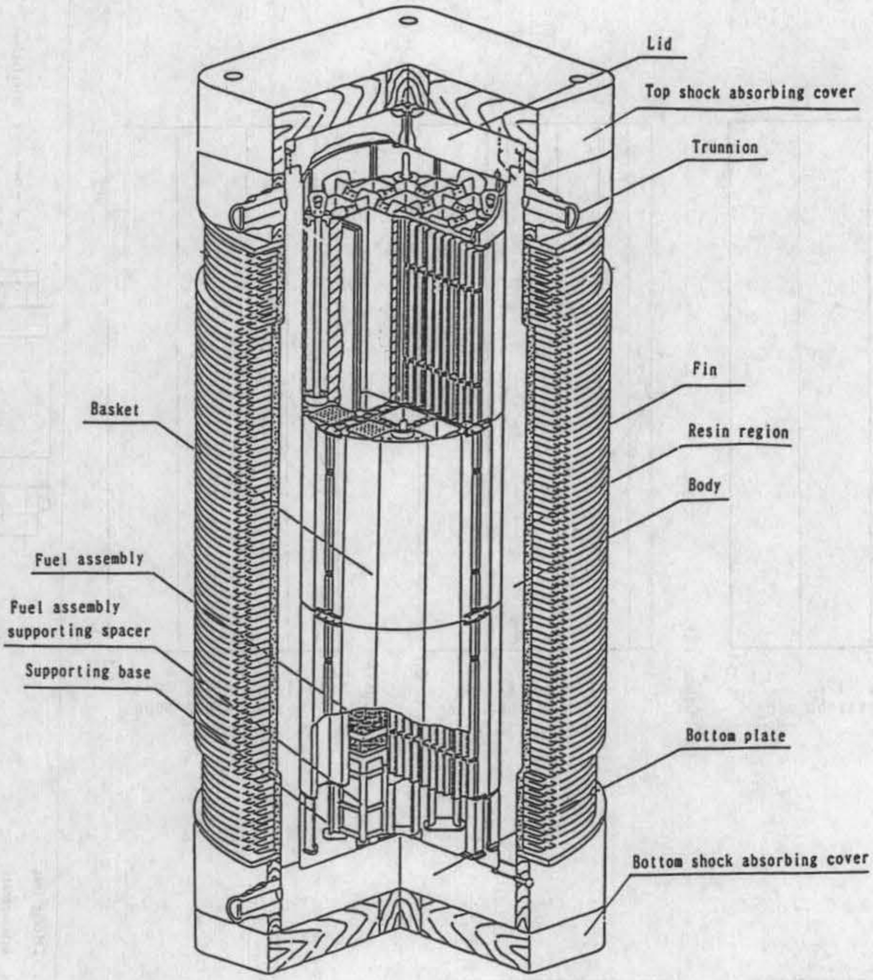

Figure 1 Illustration of TN-12/1 packaging

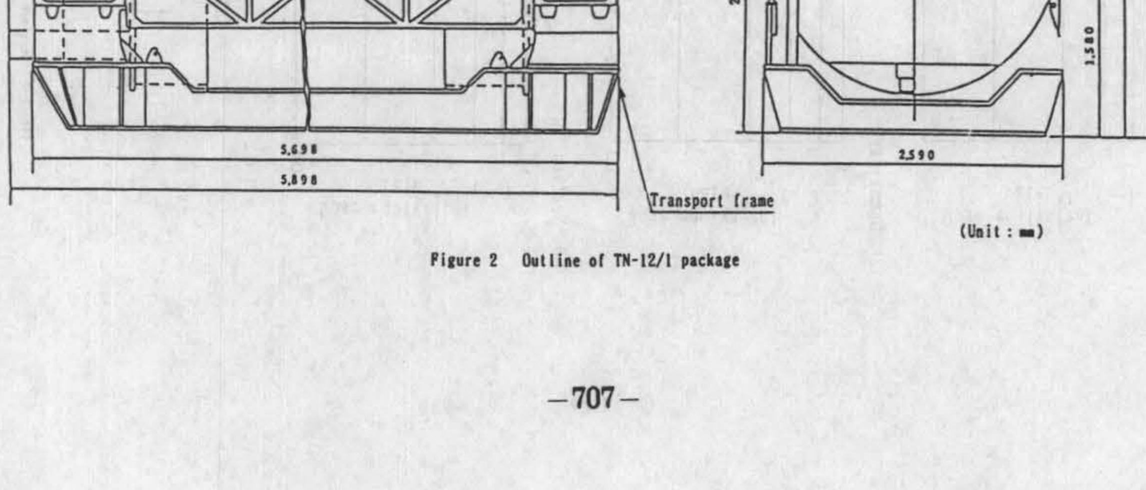

Figure 2 Outline of TN-12/1 package

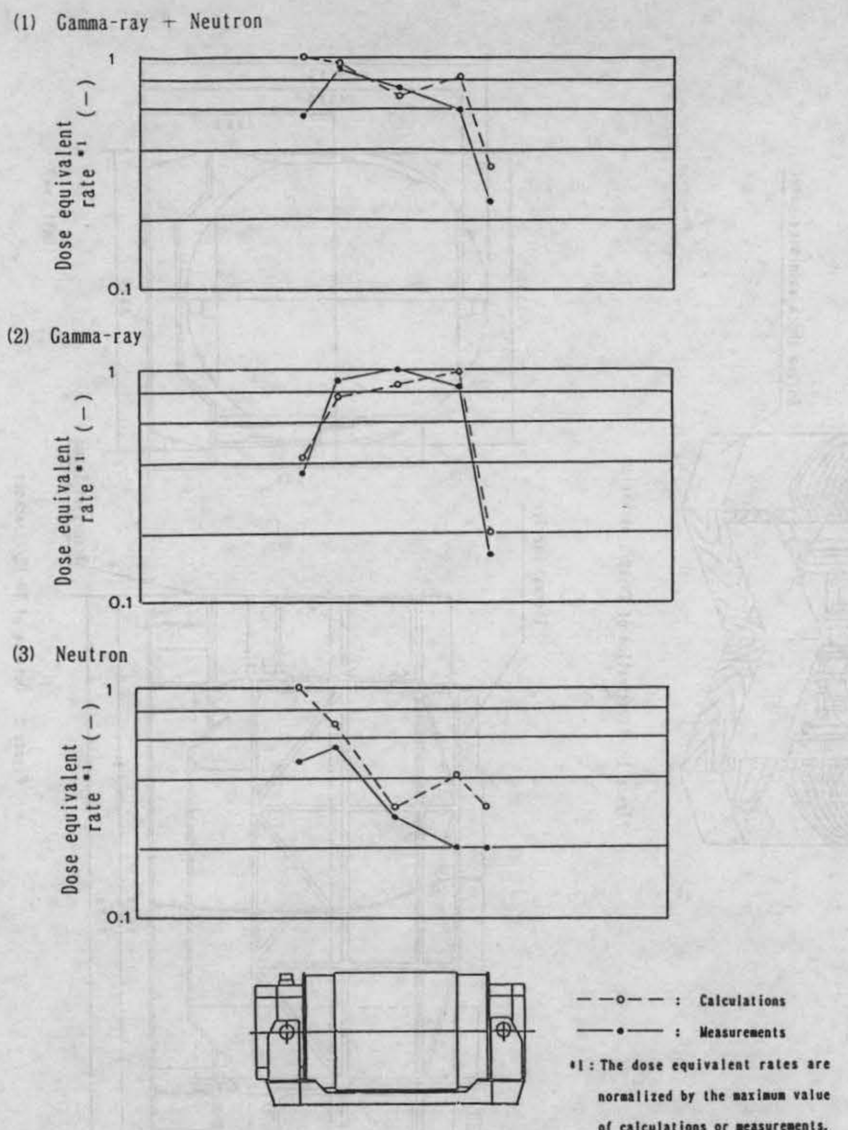

Figure 3 Sureface dose equivalent rate in the longitudinal direction

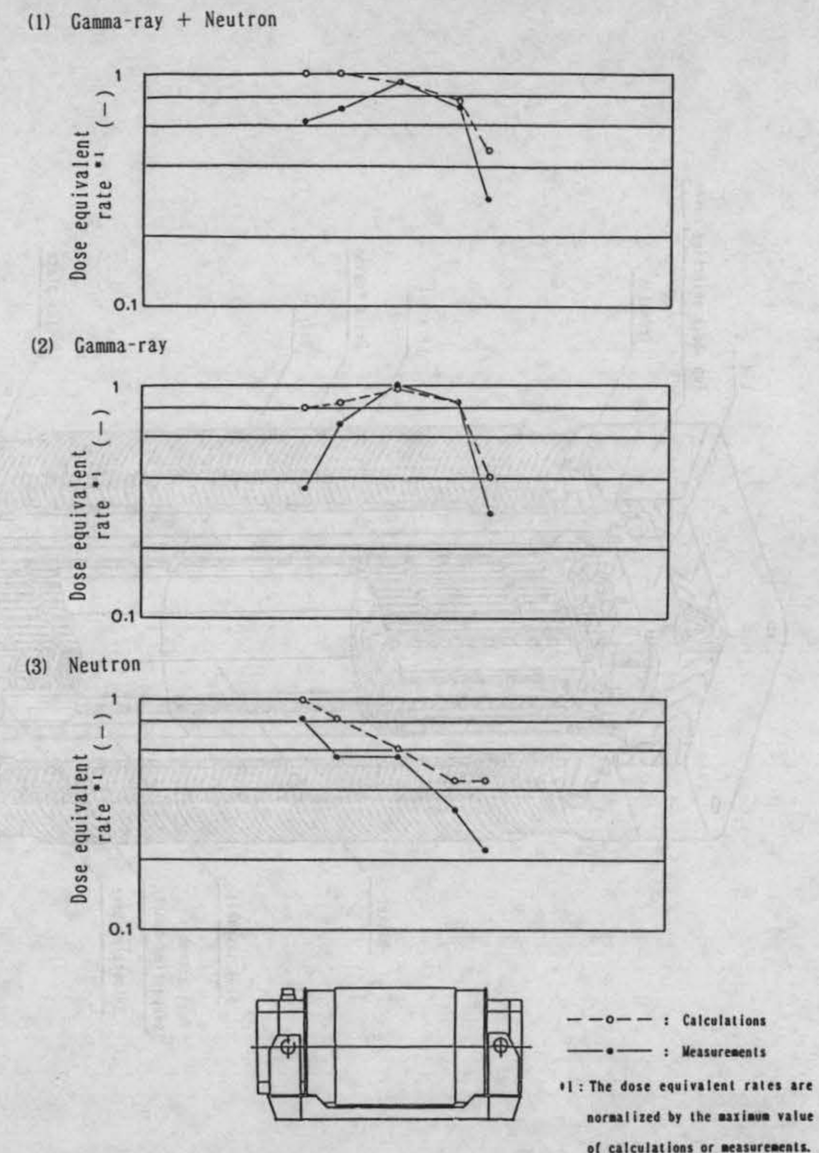

Figure 4 Dose equivalent rate in the longitudinal direction at 1m from the surface

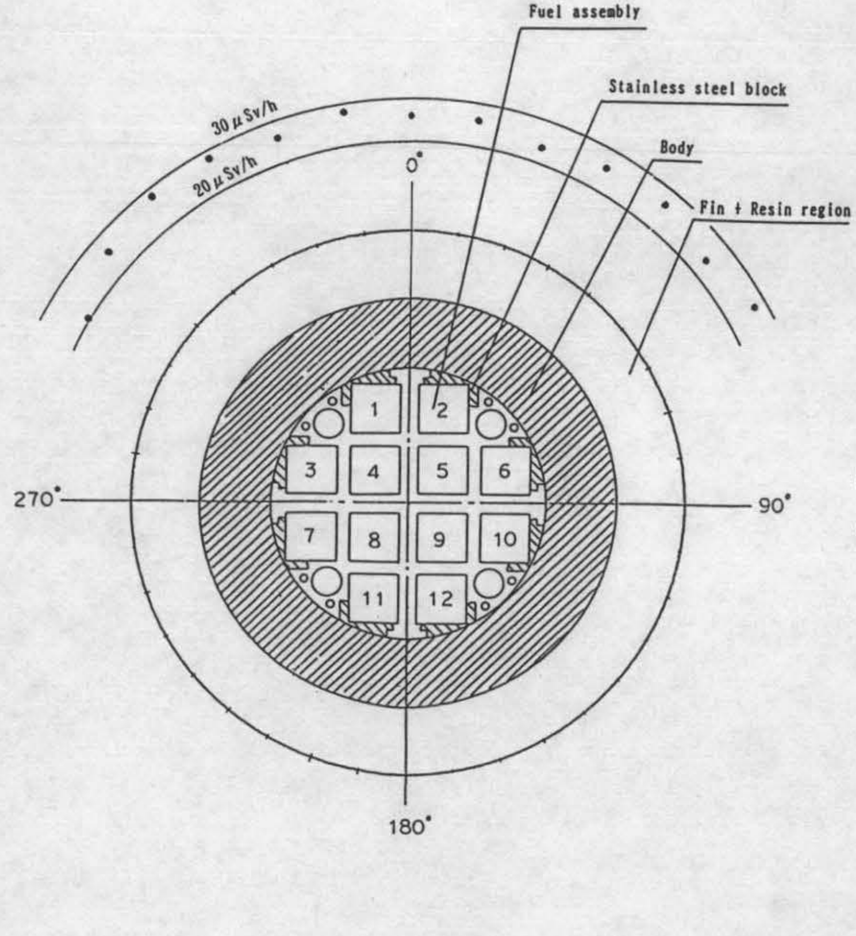

Figure 5 Gamma-ray dose equivalent rate distribution in the circumferential direction
(At 1m from the surface)

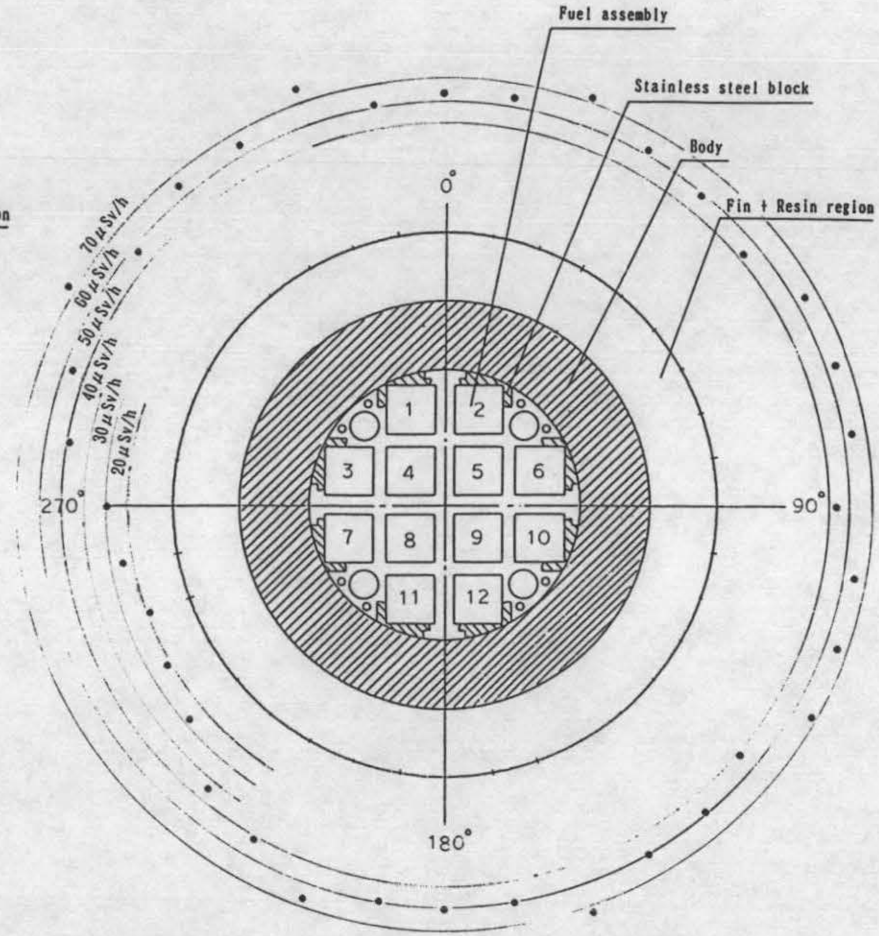

Figure 6 Gamma-ray dose equivalent rate distribution in the circumferential direction
(Surface)

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