

Development of cross section SFCX-J33 for spent fuel transport cask shielding calculation

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1. Abstract

The transport cask shielding analysis in Japan is carried out using the 2-dimensional Sn transport calculation code DOT²⁾ and cross section set DLC-23/CASK³⁾. Cross section set DLC-23/CASK was created for transport cask shielding calculation in 1973. Since this cross section set is user-friendly, it has been widely used in the shielding evaluation relevant to a spent fuel transport cask, and it has many track records.

On the other hand, it was comparatively old aged, and has been pointed out that estimation accuracy is reduced in the specific condition such as particles penetrate in iron. The main reason why DLC-23/CASK has been long used as a standard cross section for transport cask in Japan is that we had no other more appropriate cross section data set for transport cask shielding calculation.

For this reason, based on the newest evaluated nuclear data library in Japan, cross section data set SFCX-J33 for transport cask shielding calculation was developed. SFCX-J33 is the cross section set that improved the defects of DLC-23/CASK mentioned above. The main improvements are as follows.

- (1) The number of neutron energy groups was increased to 50 groups from 22 groups of DLC-23/CASK for the improvement in estimation accuracy.
- (2) The JENDL-3.3 developed by Japan Atomic Energy Agency was used as the evaluated nuclear data library.
- (3) SFCX-J33 was produced to be as user-friendly as DLC-23/CASK.

Since estimation accuracy needed to be verified about developed cross section data set SFCX-J33, the shielding examination was performed using the main materials used for a transport cask. The shielding experiment was conducted at the shielding experiment institution of the National Maritime Research Institute in Japan, and neutron and secondary gamma ray measurements were performed using Cf-252 as the neutron source. Verification of SFCX-J33 was carried out by performing experiment benchmark calculation using the shielding experiment result.

2. Neutron and gamma shielding experiment

The shielding experiment was conducted using the source of ²⁵²Cf spontaneous fission neutron.

The neutron intensity used for ²⁵²Cf source is 1.77×10^7 (neutrons/s).

First, the slab geometry was set up and the neutron, the amount of gamma rays, and the neutron spectrum were acquired.

Next, data on the amount of a neutron and gamma rays were obtained from the similar experiment with the cask geometry which imitated the actual transport cask.

These experimental results are compared with shielding calculation results in Chapter 3.

2.1 Experiment of slab geometry

The slab geometry consists of a radiation source pit and a plurality of shielding slabs. Radiation source pit is formed of a paraffin cube of 50 cm on each side, having a conical opening of 25 cm in diameter. The ²⁵²Cf locating at the center of the cube irradiates neutron towards the shielding slabs. The paraffin cube is for preventing fast neutrons escaped from the radiation source pit from becoming scattered rays and reaching the detector taking the long way around the shield to be counted as the amount of radiation. Shielding slabs used in the experiment are made of a stainless steel (SUS 304, this will be referred to simply as SUS), resin, polyethylene and lead. Dimensions of the slabs are 80 cm x 80 cm x 5 cm (T), except lead slab; while lead slab has a size of 50 cm x 50 cm x 5 cm (T). These shielding slabs are placed adjusting their centers to the straight line which connects the detector and the radiation source. Fig. 2-1 shows the structural outline of slab geometry.

Up to four shielding slabs were used at a time; when less than four shielding slabs were used, they

were placed in the order beginning with the position closest to the radiation source pit, ref. Fig. 2-1. The shielding slab disposition will be described starting from the one which is disposed to be closest to radiation source; for example, “SUS5 cm + SUS5 cm + resin 5 cm” means SUS is disposed at location 1, SUS at location 2 and resin at location 3. In the present experiment, a NaI scintillation survey meter, a rem counter, a bonner ball were used as a detector; these were installed at 133.7 cm, 133.7 cm and 101.4 cm, respectively, from the radiation source taking such factors as detection efficiency into consideration.

These ten kinds of shielding object arrangement are shown in Table 2-1.

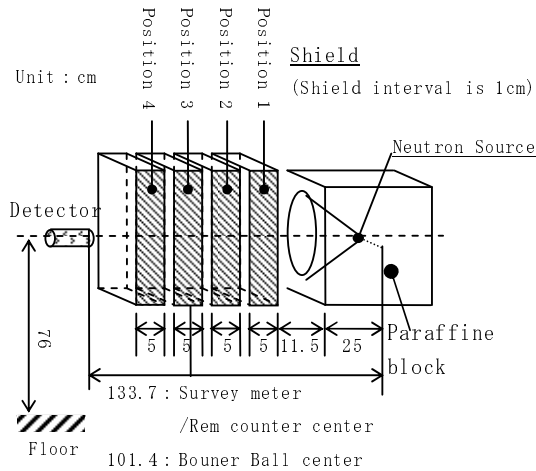


Fig. 2-1 Experiment of slab geometry

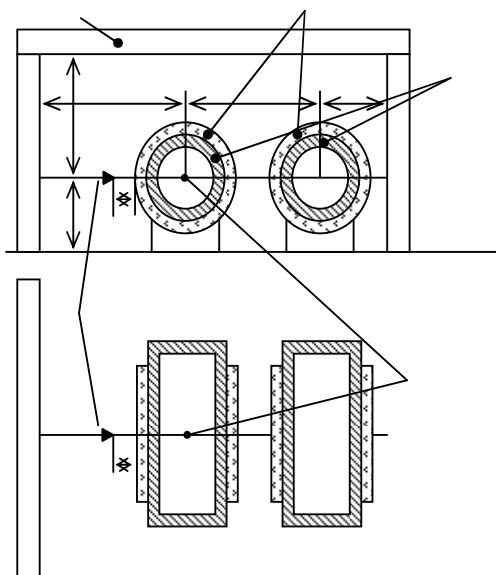


Fig. 2-2 Experiment of cask geometry

Table 2-1 Shielding pattern and Measurement value

CASE	Shielding pattern				Gamma dose rate	Neutron dose rate	Neutron spectrum
	Position 1	Position 2	Position 3	Position 4			
1-1	SUS				○	○	○
1-2	SUS	SUS			○	○	○
2	RE				○	○	○
3-1	SUS	RE			○	○	○
3-2	SUS	SUS	RE		○	○	○
4	SUS	LEA D	SUS	RE	○	○	
5-1	PE				○	○	○
5-2	PE	PE			○	○	
6-1	SUS	PE			○	○	○
6-2	SUS	PE	PE		○	○	

SUS : SUS304、RE : Resin、PE : Polyethylene

2.2 Experiment of cask geometry

The experiment which imitated the transport cask was set up and the radiation shielding experiment with ²⁵²Cf was conducted. The shielding object of a transport cask consists of resin and carbon steel. The outline of cask geometry is shown in Fig. 2-2

3. Verification by benchmark calculation of SFCX-J33

3.1 SFCX-J33 for transport cask shield evaluation

SFCX-J33 is the cross section for transport cask shielding evaluation developed by original, Nuclear Fuel Transport Inc., Mitsubishi Heavy Industries, Inc., and Mitsubishi Research Institute, Inc.

The development target of SFCX-J33 is shown in Table 3-1. It was set up from improvement in evaluation accuracy, and a viewpoint of convenience, etc. Cross section processing was created on the conditions shown in Table 3-2 using NJOY⁴⁾ and TRANSX⁵⁾. In shielding evaluation of a transport cask, iron material, such as carbon steel stainless steel, influences neutron shielding evaluation accuracy greatly. For this reason, about elements, such as Fe with the large influence of resonance, Cr, and nickel, the cross section was processed in consideration of the self-cover factor for every component, such as stainless steel (SUS304) and carbon steel. The processing flow of a cross section is shown in Fig.3-1.

Table 3-1 Development target of SFCX-J33

Development target	The spec of SFCX-J33
Reflection of the newest nuclear data	Neutron and 2nd gamma production – JENDL3.3
	Gamma – ENDF/B-VI (EPDL-97)
Improvement in accuracy of neutron shielding calculation (iron material specially)	The group of a neutron is established to be fine at an iron resonance region. Neutron 50 Group、Gamma 18 Group
General improvement in accuracy	Legendre coefficients is P5
	Contraction of UP-SCATTER
Maintenance of convenience	Microscopic cross section of DOT (ANISN) form
	The group structure of a gamma ray is the same as the existing DLC-23/CASK.

Table 3-2 SFCX-J33 Creation conditions

Item	setting
code	NJOY99.112
Nuclear data	JENDL-3.3
Accuracy of Pointwise data	0.1 %
temperature	300K
Waiting spectrum	Neutron:Fission+1/E+Maxwell Gamma:Flat
Multi-group library processing code	TRANSX-2.15
Self shielding factor	1.0E+10,1.0E+6,1.0E+5, 1.0E+4, 1000 100, 10, 1, 0.1
Scattering Data	free gas model. Only Hydrogen ENDF/B-VI Thermal Scattering Data
Output type	ANISN Binary
Collapsed group	Neutron 50 group Gamma 18 group

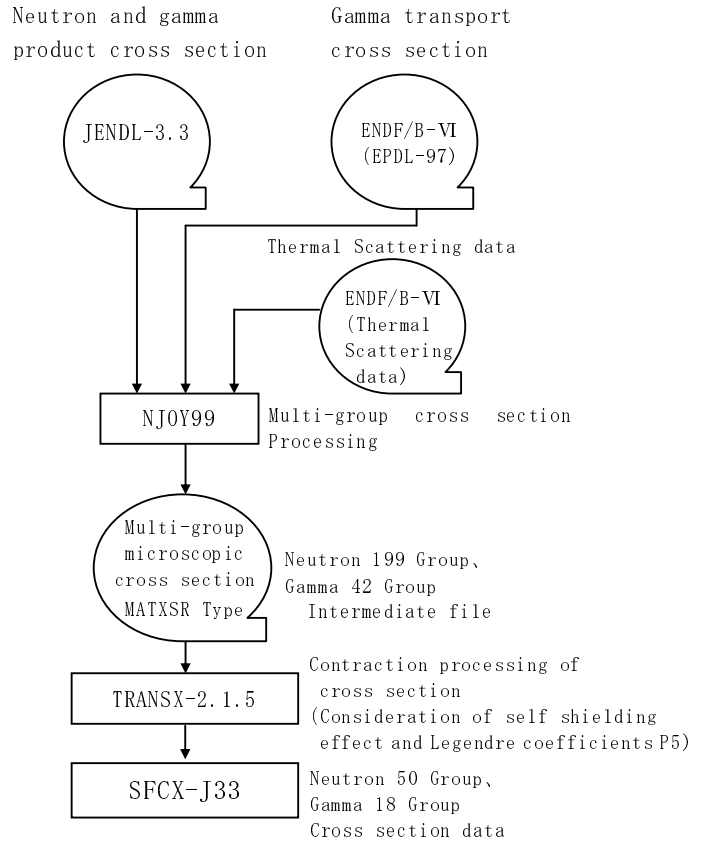


Fig.3-1 The flow of SFCX-J33 cross section

3.2 Verification result

Verification analysis was made using the new cross section SFCX-J33 based on data-processed experimental results. The analysis was conducted in combination with two-dimension transportation calculation code DOT3.5 which is used in shielding analysis for official approval of transport cask.

3.2.1 Verification result of slab geometry

Table 3-3 shows conditions for verification analysis with the slab geometry. The experiment system includes a space (air layer) between radiation source and shielding slab. DOT code used in experimental analysis tends to cause a phenomenon called a ray effect in the space region calculation which results in a deterioration in the calculation accuracy. In order to suppress the deterioration in calculation accuracy at the space, the present verification analysis was conducted with DOT's First Collision Source option (FCS option). Table 3-7

shows comparison of the experimental value (E) and the calculated value (C) with the slab geometry.

The C/E of neutron is 0.98-1.19, and that of gamma ray is 0.86-1.14, which means the values in experimental system correspond to those in calculation, with the exception of case 4 of gamma ray(SUS + lead + SUS + resin). The gamma ray in the case 4 contains lead with an extremely high gamma shield capability in experimental system; so, gamma ray reaching the detector and permeating through the shield is limited. Meanwhile in the calculation, influence of those gamma rays scattered from floor concrete, etc. and reaching the detector was not taken into account. Therefore, the gamma ray in case 4 is undervalued as C/E=0.4 in the calculation.

3.2.2 Verification result of cask geometry

Table 3-8 shows conditions for verification analysis with the cask geometry. The present cask geometry also has a space between the radiation source and the shield. Like the slab geometry, the verification was conducted using DOT's First Collision Source option. Since the experiment system is placed in a chamber formed of iron and polyethylene materials, it can be affected by the scattered rays from the wall surface. Therefore, a simulated wall surface was disposed around the cask.

Fig.3-5 shows results of dose rate analysis with gamma ray and neutron. Point of the C/E measurement is at the middle of a side. It is known from Fig.3-5 that the ray effect at the middle of the barrel has been suppressed as the result of application of FCS.

Results of comparison between the experimental value (E) and the calculated value (C) are summarized in Table 3-10. The neutron C/E is 1.01, while that of gamma ray is 1.06; a good coincidence is shown.

3.2.3 Neutron Spectrum Verification result

Neutron spectrum was measured in the same method as used in 3.2.1 at bonner ball location, the result of which was compared with experimental value-based neutron spectrum.

Fig.3-3 shows the results of comparison, in which trends can be summarized as follows:

- Experimental value (E) and calculated value (C) coincide well in the form of spectrum.
- Observing in details, measured value tends to

become higher at 1MeV or lower.

- Although there was no significant difference in the dose rate except the gamma ray in case 4 (the system containing lead), there is a little difference in the relatively low energy. This is considered to have been brought about by the fact that such ray components scattered by floor, etc. are not taken into account in the calculation.

When observed more in detail, although some differences are observed in energy bands because of restriction due to two-dimension calculation code, the spectrum provided by the new cross section SFCX-J33 corresponds to the measured results. So, it may be concluded that the calculation results using the new cross section are appropriate.

Table 3-3 Calculating Conditions for slab geometry

Item	Condition for calculation
code	DOT3.5 (FCS Option applied* 1)
Cross section	SFCX-J33 (50 Neutron-group, 18 Gamma-group)
Sn-P1	S210-P5
Radiation source intensity	Cf252-1.77E+7[n/s] (radiation source intensity at experiments)
Spectrum	Spontaneous fission spectrum of Neutron Cf252 (Table 3-4) Gamma spectrum of gamma Cf252 (Table 3-5)
Dose conversion factor	Conversion factor for 1cm dose rate on ICRP Pub74 base
Calculation Model	Fig.3-3
Combination of shielding slabs	Case1-1 SUS5cm Case1-2 SUS10cm Case2 Resin 5cm Case3-1 SUS5cm+Resin 5cm Case3-2 SUS10cm+Resin 5cm Case4 SUS5cm+Resin 5cm +Lead 5cm+SUS5cm Case5-1 Polyethylene 5cm Case5-2 Polyethylene 10cm Case6-1 SUS5cm + Polyethylene 5cm Case6-1 SUS5cm + Polyethylene 10cm
Material density	Table 3-6

* 1: First Collision Source option is provided for easing the ray effect in a space. After an initial collision ray in the system, the option is calculated using an easy method.

Table 3-4 ²⁵²Cf Spontaneous fission spectrum

No.	Upper enrgy (MeV)	Neutron spectrum
1	1.964E+01	6.157E-04
2	1.162E+01	3.699E-03
3	9.512E+00	1.047E-02
4	7.788E+00	1.795E-02
5	6.592E+00	2.261E-02
6	5.770E+00	2.246E-02
7	5.221E+00	2.737E-02
8	4.724E+00	8.905E-02
9	3.679E+00	8.730E-02
10	3.012E+00	2.314E-02
11	2.865E+00	2.352E-02
12	2.725E+00	2.375E-02
13	2.592E+00	3.986E-02
14	2.385E+00	4.010E-03
15	2.365E+00	3.981E-03
16	2.346E+00	7.976E-03
17	2.307E+00	1.590E-02
18	2.231E+00	4.726E-02
19	2.019E+00	4.606E-02
20	1.827E+00	2.237E-02
21	1.738E+00	4.322E-02
22	1.572E+00	6.044E-02
23	1.353E+00	5.449E-02
24	1.165E+00	9.043E-02
25	8.629E-01	5.732E-02
26	6.721E-01	4.339E-02
27	5.234E-01	6.089E-02
28	2.945E-01	5.116E-03
29	2.732E-01	6.063E-03
30	2.472E-01	1.196E-02
31	1.926E-01	1.937E-03
32	1.832E-01	1.808E-03
33	1.742E-01	1.683E-03
34	1.657E-01	1.571E-03
35	1.576E-01	1.036E-02
36	9.804E-02	2.341E-03
37	8.250E-02	4.318E-04
38	7.950E-02	5.723E-03
39	2.850E-02	1.300E-04
40	2.700E-02	2.350E-04
41	2.418E-02	1.028E-03
42	3.035E-03	4.667E-05
43	4.540E-04	2.724E-06
44	7.889E-05	1.931E-07
45	2.260E-05	2.793E-08
46	8.315E-06	6.609E-09
47	2.382E-06	9.558E-10
48	8.764E-07	1.895E-10
49	4.140E-07	7.478E-11
50	1.000E-07	1.669E-12

Table 3-5 ²⁵²Cf Gamma spectrum

No.	Upper enrgy (MeV)	Gamma spectrum (Photons/s)/(n/s)
1	10	4.314E-04
2	8	2.813E-03
3	6.5	1.722E-02
4	5	1.629E-02
5	4	7.588E-02
6	3	8.446E-02
7	2.5	1.455E-01
8	2	1.708E-01
9	1.66	2.550E-01
10	1.33	3.390E-01
11	1	7.334E-01
12	0.8	3.667E-01
13	0.6	0.000E+00
14	0.4	0.000E+00
15	0.3	1.143E+00
16	0.2	5.725E-01
17	0.1	0.000E+00
18	0.045	9.457E-01

Table 3-6 Material and density (Slab & Cask geometry calculation)

Material		density(g/cm3)
Source Cover	Paraffin	0.90
Shielding material	Stainless steel (SUS304)	7.9
	Carbon steel	7.85
	Lead	11.34
	Resin	1.68
	Polyethylene	0.92
Air		0.0012

Table 3-7 Benchmark calculation result

Table 3-8 Calculation conditions
(Cask geometry)

Item	Calculation conditions
code	DOT3.5 (FCS Option applied * 1)
Cross section	SFCX-J33 (50 Neutron-group, 18 Gamma-group)
Sn-PL	S160-P5
Radiation source intensity	Cf252-1.77E+7[n/s] (radiation source intensity at experiments)
Spectrum	Spontaneous fission spectrum of Neutron Cf252 (Table 3-4) Gamma spectrum of gamma Cf252 (Table 3-5)
Dose conversion factor	Conversion factor for 1cm dose rate on ICRP Pub74 base
Calculation Model	Fig.3-3
Material And density	Table 3-10

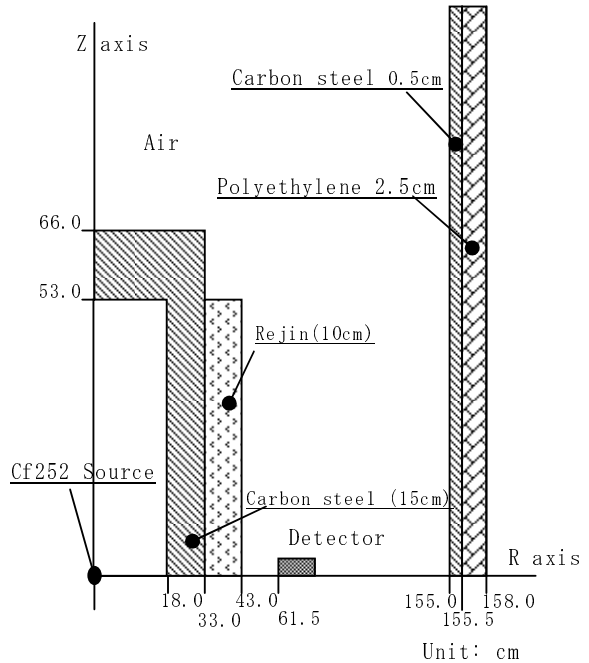


Fig.3-4 Cask geometry calculation model

Table 3-9 Material and density
(Cask geometry)

Material	density(g/cm3)
Carbon steel	7.85
Resin	1.68
Polyethylene	0.92
Air	0.0012

Table 3-10 Benchmark calculation result
(Cask geometry)

Item	Dose rate [microSv/h] , □=C/E			
	Neutron	Gamma		
		2nd Gamma	²⁵² Cf Gamma	Total
Calculation	28.8	2.49	0.19	2.68
Experiment	28.4	2.52		
C/E	1.01	1.06		

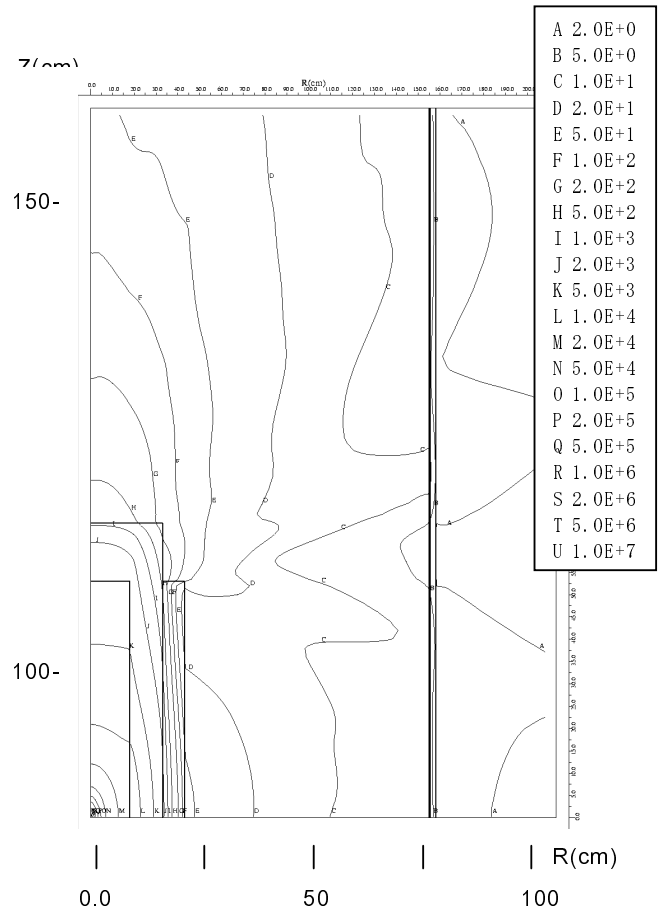


Fig.3-5 Calculation result of cask geometry
(Neutron)

4. Conclusion

A ^{252}Cf neutron and instantaneous secondary gamma ray shielding permeation experiment was conducted with slab geometry and cask geometry container simulating a cask. In the slab geometry, we measured dose rate of neutron and gamma ray, and neutron spectrum; while in the transport cask geometry, dose rate of neutron and gamma ray were measured. In the neutron spectrum measurement, we succeeded in obtaining a spectrum that has a high physical precision.

Based on the shield permeation experiment, an experimental benchmark calculation was made using the new cross section (SFCX-J33). Regarding the results of verification calculation of 1cm dose rate, experimental results corresponded to calculation using the new cross section in both the neutron and the gamma ray, with the exception of some cases where scattered rays greatly affect the dose rate.

In the neutron spectrum, spectrum shape based on calculation with the new cross section corresponds to that obtained by unfolding the experimental values. The new cross section (SFCX-J33) was verified using the experimental data compared with dose rate and the spectrum. The verification results confirmed that the new cross section provided satisfactory results and good conformity with validation of dose rate for the transport cask.

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