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Low-Activation Reinforced Concrete Design Methodology (2)
- Multi-group X-Sec. Library for Precise Activation Analysis -

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ABSTRACT

Precise estimation method for residual radioactivity and decommissioning cost is indispensable to decide adoption of low-activation material. For this precise estimation, both accurate estimation of thermal neutron flux and accurate activation cross section of structural material is necessary. In this paper, we show new cross section library that has multi-group structure in thermal energy. These libraries are processed from evaluated cross section library JENDL 3.3 by using NJYOY 99.83. Cross section library for S_N transport calculation and activation calculation are prepared. These libraries are tested by JPDR experiment, and found to give better result than single thermal group treatment.

INTRODUCTION

LARC (Low-Activation Reinforced Concrete) project aims to reduce residual radioactivity of concrete under the clearance level at decommissioning phase of nuclear power plant. This project consists from three parts; namely, development of low activation reinforced concrete material, development of impurity database in raw materials, and developments of precise and user friendly radioactivity estimation system to decide region of low-activity material.

To perform precise radioactivity estimation, we need precise neutron flux estimation and precise activation estimation. We now have two major methods to estimate neutron flux; Monte Carlo method and Discrete Ordinates S_N method. Monte Carlo method is useful because of its ability of precise geometry description and continuous energy treatment. However, this method is time-consuming to estimates neutron flux distribution in thick biological shield wall (BSW). There are many variance reduction methods to save computer time. But if we use strong variance reduction method, we encounter difficulty to make out reliability of obtained results. To avoid this, we decided to use Discrete Ordinates S_N method.

Discrete Ordinates S_N method is useful to estimate neutron flux distribution for big geometry, but its geometry description ability is limited because it uses spatial mesh. However it is not serious for bulk shielding problem. Second problem arises from energy mesh. Because discrete Ordinates S_N method uses group-wise neutron cross-section library, it is necessary to use adequate energy group number. MATXSLIB-J33 [1] is one of the precise libraries that energy group number is 175.

Because almost the neutron reactions to produce residual radioactivity occurs at thermal energy, it is necessary to treat thermal neutron precisely. From this point of view, even MATXSLIB-J33 library is not enough because thermal energy group is only two. In this work, we developed new group wise cross section libraries with ten thermal neutron groups. One is transport cross-section library, which is used in Discrete Ordinates S_N method to perform neutron flux calculation. The other is activation cross-section library, which is used to calculate radioactivity with neutron flux data.

METHOD

Cross Section Library for Transport Calculation

We processed JENDL3.3 [2] evaluated cross-section library to get multi-group transport cross-section library by using NJOY 99.83[3]. Number of neutron energy groups is 183 (thermal neutron energy groups is 10) and number of photon energy groups is 42. Number of nuclides processed is 337.

Figure 1 shows process flow to make 183-group library. Table 1 shows parameters used in this process. The difference between MATXSLIB-J33 library and this library is only number of thermal groups.

Figure 2 shows ^{56}Fe total cross-section as a sample. Thin red line shows JENDL 3.3 continuous energy cross-section. Processed 183-groups library is shown in thick black histogram. Dotted histogram shows 175-groups MATXS-J33 library. We can see that the 183-groups

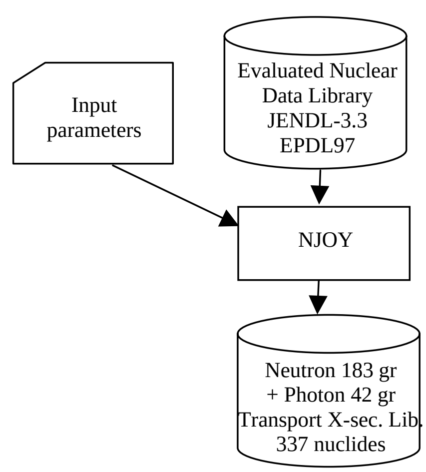

Fig.1 Process Flow Process Flow of Multi-group transport cross-section Library

library is same as 175-groups MATXS-J33 library except thermal region, which is more precise in the 183-groups library.

Table 1 parameters for transport cross section library

Code	NJOY99.83
Evaluated nuclear data library	JENDL3.3 [3] [neutron] EPDL97 [4] [photon]
Accuracy of point-wise data	0.1%
Temperature	300K
Group structure	183 groups neutron [10 groups thermal neutron] 42 groups photon
Upper energy of thermal treatment	4.6eV
Process code for multi-group library	TRANSX-2.15
Thermal scattering data	Free gas model [S(α, β) for hydrogen in water]
Self shielding factor	10^{10}, 10^4, 10^3, $3x10^2$, 10^2, $3x10^1$, 10^1, 10^0, 10^{-1}, 10^{-5}
Number of Legendre expansion	P_6

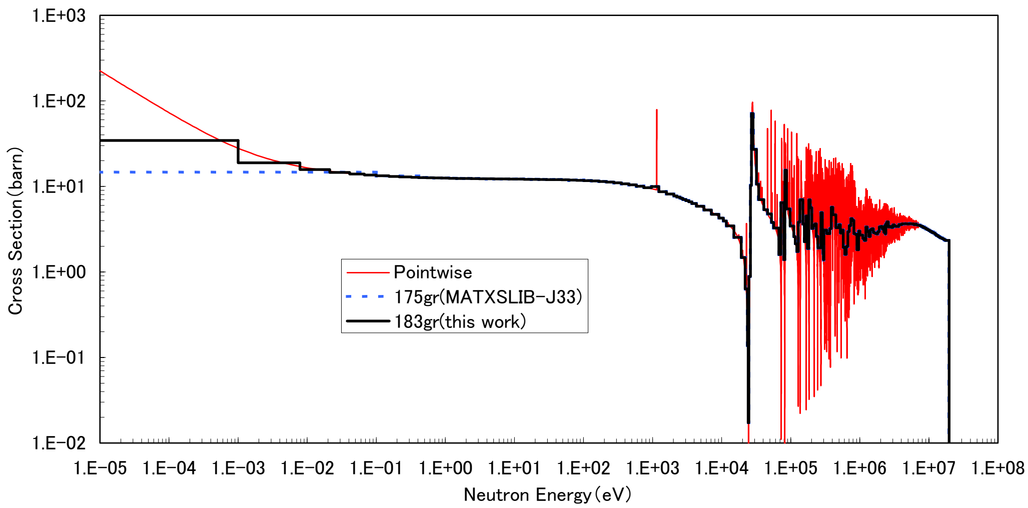

Fig.2 ^{56}Fe total cross-section in 183gr library

Cross Section Library for Activation Calculation

We processed JENDL3.3 [2] evaluated cross-section library to get multi-group activation cross-section library by using NJOY 99.83[3]. Number of neutron energy groups is 183 (thermal neutron energy groups is 10). Number of nuclides processed is 105, which is important in decommissioning phase. Reactions processed are (n, 2n), (n, γ), (n, p) and (n, α). Basically, we used JENDL3.3 library, but we used JENDL activation file [5] when nuclide is not included in JENDL3.3. Table 2 shows nuclides and reactions processed for activation library.

Figure 3 shows process flow to make 183-group library. Table 3 shows parameters used in this process.

Figure 4 and 5 show ^{59}Co(n, γ) ^{60}Co and ^{151}Eu(n, γ) ^{152}Eu cross-section, respectively. Thin red line shows JENDL 3.3 continuous energy cross-section. Processed 183-groups library is shown in thick black histogram. Dotted histogram shows 175-groups cross-section. We can see that the 183-groups library is same as 175-groups MATXS-J33 library except thermal region, which is more precise in the 183-groups library.

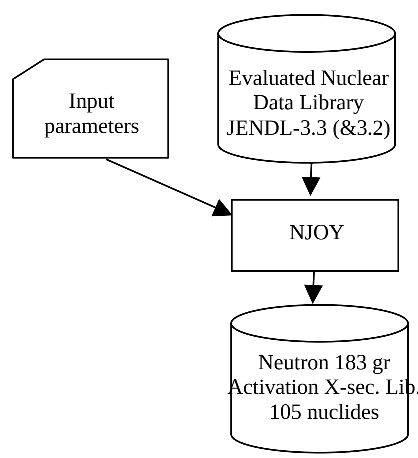

Fig.3 Process Flow of Multi-group activation cross-section Library

Table 2 Processed activation cross-sections (Nuclide and reaction)

Nuclide		Reaction				Nuclide		Reaction			
1	H-2	(n,2n)	(n,γ)			54	Sb-126			(n,p)	
2	He-3		(n,γ)	(n,p)		55	Te-122	(n,2n)	(n,γ)	(n,p)	(n,α)
3	Li-6	(n,2n)	(n,γ)	(n,p)	(n,α)	56	Te-124	(n,2n)	(n,γ)	(n,p)	(n,α)
4	C-13	(n,2n)	(n,γ)	(n,p)	(n,α)	57	Te-125	(n,2n)	(n,γ)	(n,p)	(n,α)
5	N-14	(n,2n)	(n,γ)	(n,p)	(n,α)	58	I-127	(n,2n)	(n,γ)	(n,p)	(n,α)
6	O-17	(n,2n)		(n,p)	(n,α)	59	I-129	(n,2n)	(n,γ)	(n,p)	(n,α)
7	Na-23	(n,2n)	(n,γ)	(n,p)	(n,α)	60	Xe-129	(n,2n)	(n,γ)	(n,p)	(n,α)
8	Mg-24		(n,γ)	(n,p)	(n,α)	61	Cs-133	(n,2n)	(n,γ)	(n,p)	(n,α)
9	Al-27	(n,2n)	(n,γ)	(n,p)	(n,α)	62	Cs-134	(n,2n)	(n,γ)	(n,p)	(n,α)
10	Cl-35	(n,2n)	(n,γ)	(n,p)	(n,α)	63	Cs-135	(n,2n)	(n,γ)	(n,p)	(n,α)
11	Cl-37	(n,2n)	(n,γ)	(n,p)	(n,α)	64	Cs-136	(n,2n)	(n,γ)	(n,p)	(n,α)
12	Ar-36				(n,α)	65	Ba-132	(n,2n)	(n,γ)	(n,p)	(n,α)
13	K-39	(n,2n)	(n,γ)	(n,p)	(n,α)	66	Ba-134	(n,2n)	(n,γ)	(n,p)	(n,α)
14	Ca-40		(n,γ)	(n,p)	(n,α)	67	Ba-137	(n,2n)	(n,γ)	(n,p)	(n,α)
15	Ca-42	(n,2n)	(n,γ)	(n,p)	(n,α)	68	Sm-150	(n,2n)	(n,γ)	(n,p)	(n,α)
16	Sc-45	(n,2n)	(n,γ)	(n,p)	(n,α)	69	Sm-151	(n,2n)	(n,γ)	(n,p)	(n,α)
17	Ti-46	(n,2n)	(n,γ)	(n,p)	(n,α)	70	Sm-152	(n,2n)	(n,γ)	(n,p)	(n,α)
18	Mn-55	(n,2n)	(n,γ)	(n,p)	(n,α)	71	Sm-154	(n,2n)	(n,γ)	(n,p)	(n,α)
19	Fe-54	(n,2n)	(n,γ)	(n,p)	(n,α)	72	Eu-151	(n,2n)	(n,γ)	(n,p)	(n,α)
20	Fe-56	(n,2n)	(n,γ)	(n,p)	(n,α)	73	Eu-152	(n,2n)	(n,γ)	(n,p)	(n,α)
21	Fe-58	(n,2n)	(n,γ)	(n,p)	(n,α)	74	Eu-153	(n,2n)	(n,γ)	(n,p)	(n,α)
22	Co-59	(n,2n)	(n,γ)	(n,p)	(n,α)	75	Eu-154	(n,2n)	(n,γ)	(n,p)	(n,α)
23	Ni-58	(n,2n)	(n,γ)	(n,p)	(n,α)	76	Eu-155	(n,2n)	(n,γ)	(n,p)	(n,α)
24	Ni-60	(n,2n)	(n,γ)	(n,p)	(n,α)	77	Gd-152	(n,2n)	(n,γ)	(n,p)	(n,α)
25	Ni-62	(n,2n)	(n,γ)	(n,p)	(n,α)	78	Gd-154	(n,2n)	(n,γ)	(n,p)	(n,α)
26	Ni-64	(n,2n)	(n,γ)	(n,p)	(n,α)	79	Gd-155	(n,2n)	(n,γ)	(n,p)	(n,α)
27	Cu-63	(n,2n)	(n,γ)	(n,p)	(n,α)	80	Tb-159	(n,2n)	(n,γ)	(n,p)	(n,α)
28	Zn-64	(n,2n)	(n,γ)	(n,p)		81	Dy-160	(n,2n)		(n,p)	
29	Zn-66	(n,2n)		(n,p)	(n,α)	82	Ta-181	(n,2n)	(n,γ)	(n,p)	(n,α)
30	Sr-89	(n,2n)	(n,γ)	(n,p)	(n,α)	83	Ta-182		(n,γ)	(n,p)	
31	Sr-90	(n,2n)	(n,γ)	(n,p)	(n,α)	84	W-182	(n,2n)	(n,γ)	(n,p)	(n,α)
32	Zr-93	(n,2n)	(n,γ)	(n,p)	(n,α)	85	Re-185	(n,2n)	(n,γ)	(n,p)	(n,α)
33	Zr-94	(n,2n)	(n,γ)	(n,p)	(n,α)	86	U-232	(n,2n)	(n,γ)	(n,f)	(n,3n)
34	Zr-95	(n,2n)	(n,γ)	(n,p)	(n,α)	87	U-233	(n,2n)	(n,γ)	(n,f)	(n,3n)
35	Nb-93	(n,2n)	(n,γ)	(n,p)	(n,α)	88	U-234	(n,2n)	(n,γ)	(n,f)	(n,3n)
36	Nb-94	(n,2n)	(n,γ)	(n,p)	(n,α)	89	U-235	(n,2n)	(n,γ)	(n,f)	(n,3n)
37	Nb-95	(n,2n)	(n,γ)	(n,p)	(n,α)	90	U-236	(n,2n)	(n,γ)	(n,f)	(n,3n)
38	Mo-92	(n,2n)	(n,γ)	(n,p)	(n,α)	91	U-237	(n,2n)	(n,γ)	(n,f)	(n,3n)
39	Mo-94	(n,2n)	(n,γ)	(n,p)	(n,α)	92	U-238	(n,2n)	(n,γ)	(n,f)	(n,3n)
40	Mo-95	(n,2n)	(n,γ)	(n,p)	(n,α)	93	Np-236	(n,2n)	(n,γ)	(n,f)	(n,3n)
41	Mo-98	(n,2n)	(n,γ)	(n,p)	(n,α)	94	Np-237	(n,2n)	(n,γ)	(n,f)	(n,3n)
42	Mo-99	(n,2n)	(n,γ)	(n,p)	(n,α)	95	Np-238	(n,2n)	(n,γ)	(n,f)	(n,3n)
43	Tc-99	(n,2n)	(n,γ)	(n,p)	(n,α)	96	Np-239	(n,2n)	(n,γ)	(n,f)	(n,3n)
44	Ru-96	(n,2n)	(n,γ)	(n,p)	(n,α)	97	Pu-238	(n,2n)	(n,γ)	(n,f)	(n,3n)
45	Ru-99	(n,2n)	(n,γ)	(n,p)	(n,α)	98	Pu-239	(n,2n)	(n,γ)	(n,f)	(n,3n)
46	Ru-106	(n,2n)	(n,γ)	(n,p)	(n,α)	99	Pu-240	(n,2n)	(n,γ)	(n,f)	(n,3n)
47	Ag-107	(n,2n)	(n,γ)	(n,p)	(n,α)	100	Pu-241	(n,2n)	(n,γ)	(n,f)	(n,3n)
48	Ag-109	(n,2n)	(n,γ)	(n,p)	(n,α)	101	Pu-242	(n,2n)	(n,γ)	(n,f)	(n,3n)
49	Sn-124	(n,2n)	(n,γ)	(n,p)	(n,α)	102	Am-241	(n,2n)	(n,γ)	(n,f)	(n,3n)
50	Sn-126	(n,2n)	(n,γ)	(n,p)	(n,α)	103	Am-242	(n,2n)	(n,γ)	(n,f)	(n,3n)
51	Sb-123	(n,2n)	(n,γ)	(n,p)	(n,α)	104	Am-243	(n,2n)	(n,γ)	(n,f)	(n,3n)
52	Sb-124	(n,2n)	(n,γ)	(n,p)	(n,α)	105	Am-244m	(n,2n)	(n,γ)	(n,f)	(n,3n)
53	Sb-125	(n,2n)	(n,γ)	(n,p)	(n,α)						

(yellow)	from JENDL-3.2 Activationfile	9 nuclides
	from JENDL3.3	96 nuclides
	total	105 nuclides

Table 3 parameters for activation cross-section library

Code	NJOY99.83
Evaluated nuclear data library	JENDL3.3 JENDL Activation X-sec. File 97
Accuracy of point-wise data	0.1%
Temperature	300K
Group structure	183 groups neutron [10 groups thermal neutron]
Upper energy of thermal treatment	4.6eV
Process code for multi-group library	TRANSX-2.15
Weighting spectrum	
10^{-5} to 0.125eV	Maxwellian thermal spectrum
0.125eV to 0.8208 MeV	1/E slowing down spectrum
0.8208 MeV	Fission spectrum

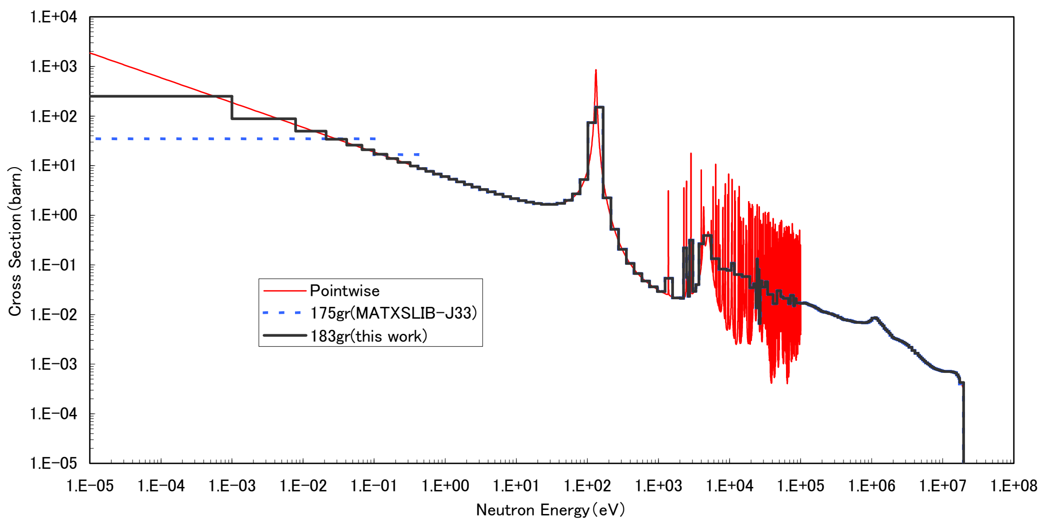

Fig.4 ^{59}Co(n, γ) ^{60}Co cross-section in 183-group library

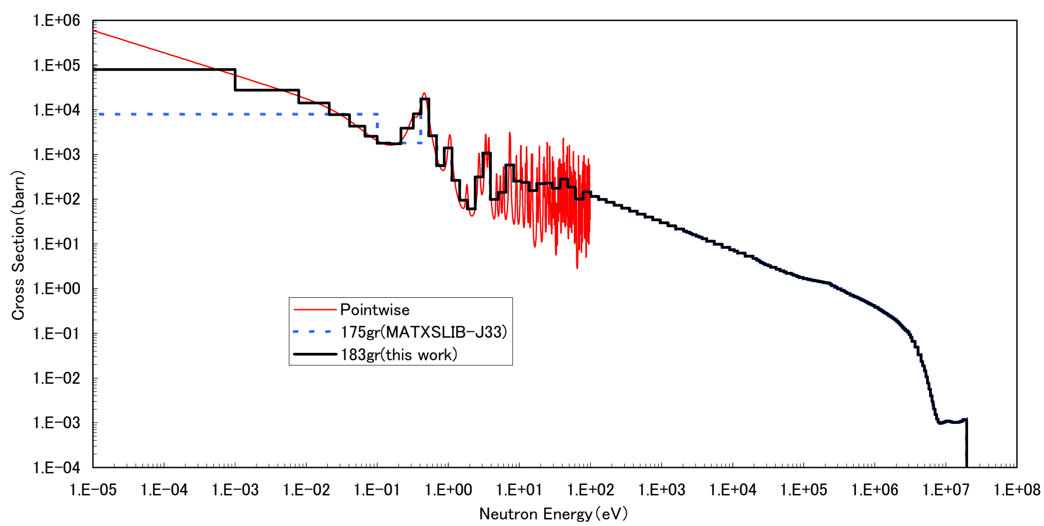

Fig.5 ^{151}Eu(n, γ) ^{152}Eu cross-section in 183-group library

Cross Section Library Test using JPDR Experiment

We tested these cross section libraries by using experimental data. Many benchmark experiments were carried out and published up to now. But the shielding experiment of thick concrete (~2m) using reactor is rare. We used JPDR experiment by Sukegawa (1993)[6] to test cross-section libraries.

Figure 6 shows calculation flow of JPDR activation analysis. We used two-dimensional S_N transport code DORT [7] with 183-group cross section library. Because this library has ten thermal energy groups, we can get correct thermal energy spectrum using “outer iteration” e.g. energy iteration. We also used conventional 175-group MATXSLIB-J33 library without using “outer iteration” for comparison. Geometry coverage is up to biological shield wall shown in Fig.7.

Figure 8 shows calculated neutron flux distribution. Left figure shows fast neutron flux (19.64MeV to 1.0026MeV), center figure shows intermediate neutron flux (1.0026MeV to 0.414eV), and right figure shows thermal neutron flux (0.414eV to 10^{-5}eV).

Figure 9 shows comparison of measured and calculated radioactivity along line A in Fig.7. The sequence of radioactivity in descending order is ^{152}Eu, ^{60}Co, ^{154}Eu, and ^{134}Cs. Because these nuclides are produced by thermal neutron, radioactivity distribution is similar to thermal flux distribution. The reason of radioactivity peak around 10-cm in depth corresponds to thermal neutron peak, which is created by slowing down of fast neutron incident to concrete. Thick solid lines show calculated results using 183-group library. Thin solid lines show calculated results using conventional 175-group MATXS-J33library. We can see that we get better results by using precise thermal neutron treatment.

Figure 10 shows the ratio of radioactivity in biological concrete shield of JPDR calculated using 183-group library and 175-group library. We can see that the precise thermal energy treatment gives about 5% bigger result at 10-cm depth, but smaller result deeper than 30-cm depth. Especially, it gives about 30% smaller result compared to conventional thermal group treatment. We found that by precise treatment of thermal energy neutron, we get smaller total activity in general.

The difference between measured and calculated radioactivity is still big even if we used precise thermal treatment. The reason we suspect is the water content in concrete. Water in concrete exist as free, combined and crystal water. Underestimation of water content sometimes occurs because it depends on sample preparation method and temperature used at that time. The water content in their report [4] is 6.8%. It seems to be lower than normal concrete. If we assume that the water content in concrete as 9%, which is normal value in well-managed concrete, we can reproduce measured radioactivity shown as dotted line in Fig. 9.

Although it is regrettable that we could not reproduce the radioactivity of experimental result when we used water content from original paper, at least we realized the tendency of radioactivity using precise thermal treatment.

Table 4 parameters for two-dimensional transport calculation

Code	DORT [two dimensional S_N transport code]
Geometry	Cylinder R-Z
GroupWise X-sec. Lib	183gr X-sec. (prepared in this work) and 175gr X-sec. (MATXSLIB-J33)
Spatial mesh number	
Radial mesh	200 mesh
Axial mesh	400 mesh
Number of Legendre expansion	P_5
Angular quadrature	96 (S_{12})
Neutron Source Strength	7.518×10^{16}(s^{-1}) at 1-MWt reactor power
Boundary condition	Maxwellian thermal spectrum
Left	Reflective
Right	Free
Top	Free
Bottom	Free
Iteration Error limit	0.001

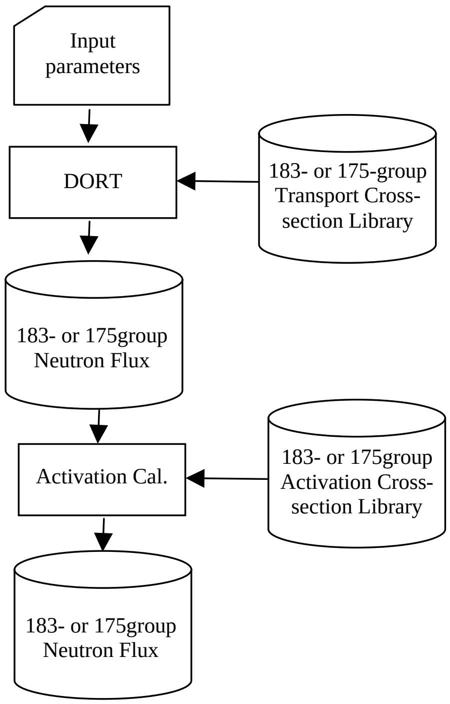

Fig.6 Flow of JPDR Activation Calculation

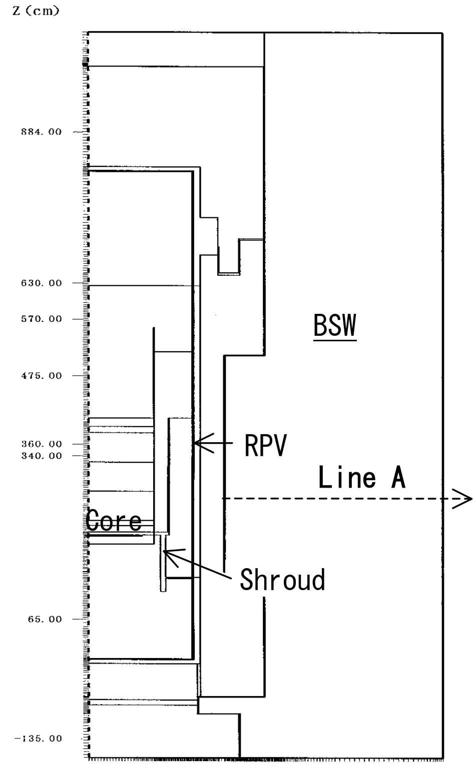

Fig.7 Geometry of JPDR for neutron transport calculation

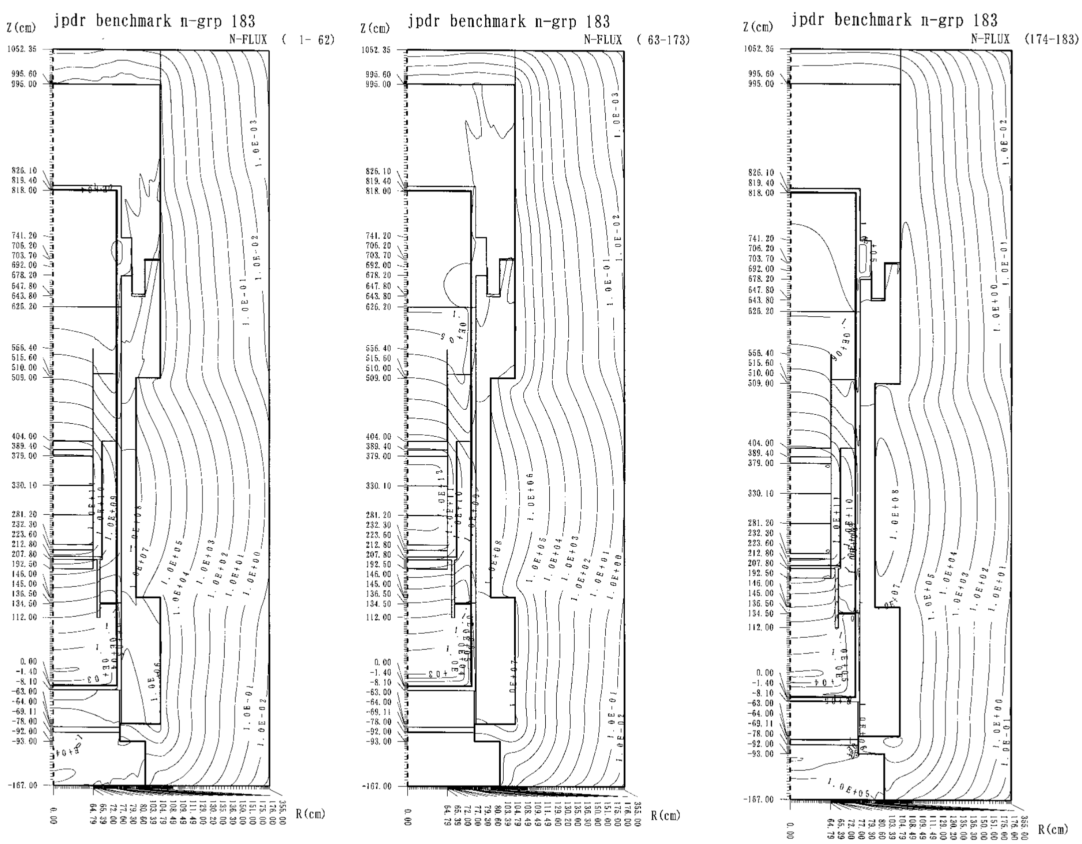

Fig.8 Calculated neutron flux distribution (Left: Fast neutron flux, Center: Intermediate neutron flux, Right: Thermal neutron flux)

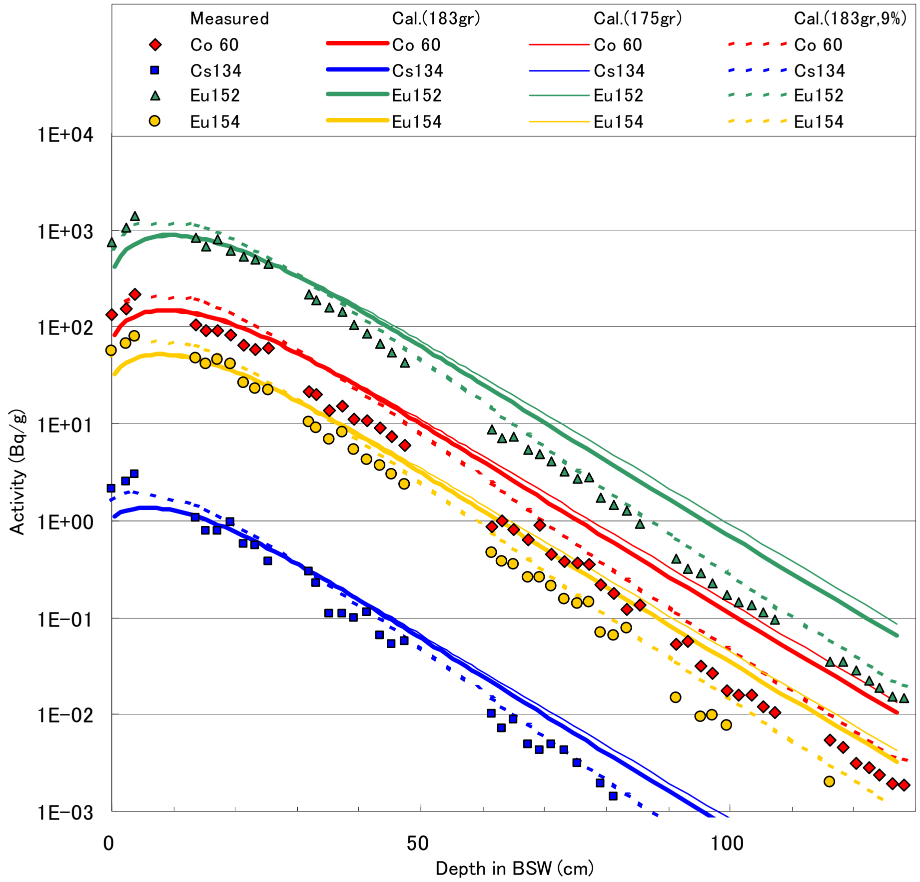

Fig. 9. Comparison of measured and calculated radioactivity

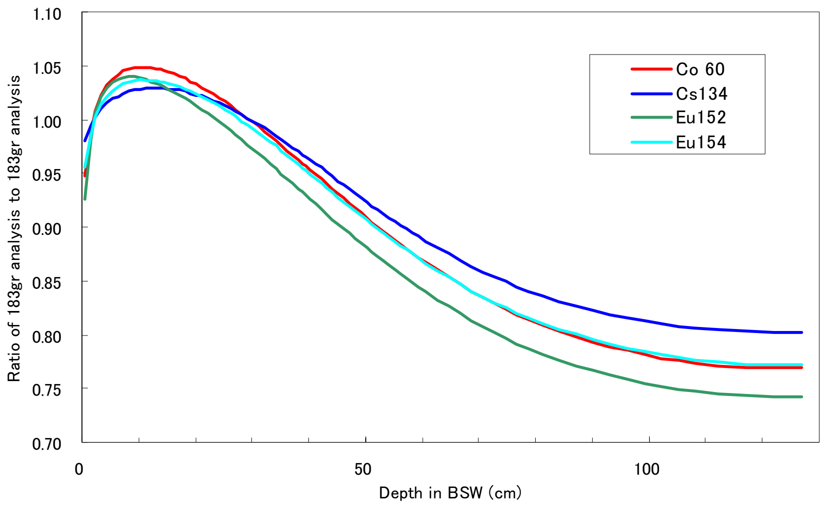

Fig. 10 Ratio of activity calculated using 183-group library and 175-group library

SUMMARY AND CONCLUSION

We have made new transport cross-section library and activation cross-section library that have 10-group structure in thermal energy. These libraries were processed from evaluated cross section library JENDL 3.3 by using NJOY 99.83. The nuclides and reactions in activation library are selected to include all the important radioactive nuclides at decommissioning phase.

These libraries are tested by JPDR experiment. From the comparison to measured data, we found that the multi-group thermal energy treatment gives better result than conventional thermal group treatment.

By using this method, we can get precise residual radioactivity and decommissioning cost, and also we can properly consider the adoption of low-activation material.

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