Study of Cross section Libraries for Shielding Design of Spent Fuel Cask and Cask Storage Facility

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ABSTRACT

The purpose of this study is to examine suitable cross section libraries for dose calculations and source term calculations for spent fuel storage casks, and discuss the applicability of respective libraries to shielding design.

Calculations were conducted by using various cross section libraries, such as DLC23/CASK, based on ENDF/B-IV, MATXSLIB-J33 based on JENDL3.3, and VITAMIN-B6 based on ENDF/B-VI. Then, the calculation results were compared with the measured values. MATXSLIB-J33 was found to be the best among these libraries for its better agreement with the measured values.

Source term calculations were conducted by using ORIGEN2&BWRU, ORIGEN2&ORLIB-J33 based on JENDL3.3, and ORIGEN-ARP&GE8x8-4. Then, the reproducibility of calculation with BWRU was confirmed and appropriate void ratios were suggested for calculations with ORLIB-J33 and GE8x8-4.

INTRODUCTION

When defining the shielding design of spent fuel storage casks and the cask storage facility, nuclear cross section library is one of the most vital factors of shielding design such as source term calculations of spent fuel, and dose calculations of bulk shielding.

Dose calculations are generally conducted by using Sn transport codes, such as DOT and DORT with DLC23/CASK in Japan, while source term calculations are conducted by using the ORIGEN codes. JENDL 3.3 and ENDF/B-VI, updated nuclear data libraries, are available to use in such calculations. However, relatively old library ENDF/B-IV is employed for these calculations even now. The applicability of the Monte Carlo codes has also been actively discussed in recent years, and greater latitude for these calculations has given. However, the application of the Monte Carlo codes requires the confirmation of validations using with new cross section libraries including comparison with the measurement. In addition, considering the peculiarities of import, understanding the differences between nuclear data libraries used by each manufacturer is also important.

The purpose of this study is to examine suitable cross section libraries for dose calculations of cask and cask storage facility and for source term calculations of spent fuel storage casks, and discuss their applicability to shielding design.

1. EXAMINATION OF CROSS SECTION LIBRARIES FOR DOSE CALCULATIONS

The dose at the surface of a dry-type cask was calculated to examine the sensitivity to the differences of cross section libraries, by using DLC23/CASK based on ENDF/B-IV, MATXSLIB-J33 based on JENDL3.3, and VITAMIN-B6 based on ENDF/B-VI [1,2,3,4,5,6]. The calculation model was a cylinder composed of 75.9 cm thick fuel, 0.6 cm thick stainless-steel, 24.6 cm thick iron, 14.3 cm thick resin, and 2 cm thick iron from the center. The calculations were classified according to source type, neutron, gamma-ray emitted from the fission product (FP-gamma), ^{239}Pu was employed as the neutron source spectrum. FP-gamma source was defined by calculated output of the ORIGEN code. The doses of a dry-type cask calculated by using the ANISN code or the Monte Carlo code MCNP5 [8, 9].

Table 1 below lists the relative doses at the surface of a dry-type cask, when setting doses as calculated by using MCNP5 with mcplib02 as criteria.

Table 1. Relative doses at the surface of a dry-type cask (mcplib02 = 1)

	MCNP5		ANISN-JR		
	mcplib02 (JENDL3.3)	ENDF/B-VI	MATXLIB-J33	VITAMIN-B6	DLC23/CASK
Neutron	**1.00**	1.07	1.01	0.91	**0.85**
FP-gamma	**1.00**	1.00	0.84	0.81	**1.12**
Total	**1.00**	1.05	0.95	0.88	0.96

Due to its higher reproducibility of MCNP5, gamma dose calculated by using DLC23/CASK could be overestimated and the neutron dose underestimated. The calculation terms should be simplified to define the characteristic differences of each library. In this study, the source was classified by gamma-ray and neutron, with the most influential material being selected for each source, and then, dose was compared with benchmark test results.

1.1 Examination for gamma ray

Figure 1-1 shows the dose attenuation in cask emitted from ^{18}F calculated by the ANISN code and Fig.1-2 shows that emitted from ^{60}Co. The calculation model was the same cask as described above. The attenuation rate in resin was negligible small against that in iron as shown in both Fig.1-1 and Fig.1-2. Therefore, iron was selected as the object of examination.

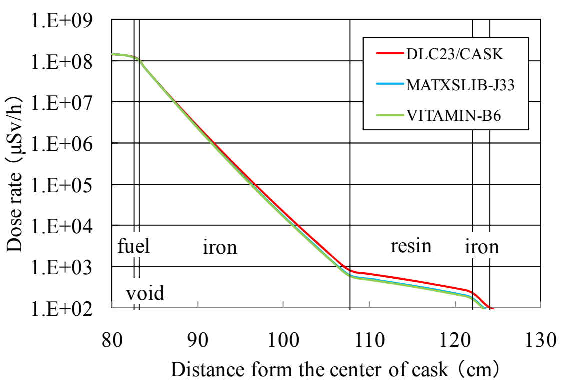

Fig. 1-1. Dose attenuation in cask emitted from ^{18}F

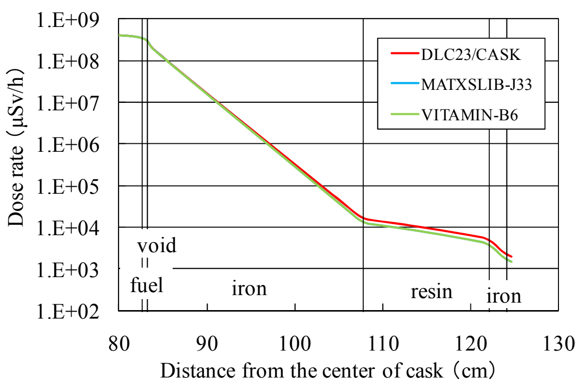

Fig. 1-2. Dose attenuation in cask emitted from ^{60}Co

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The gamma doses in iron were calculated by the sphere model having only point source and shielding to examine the attenuation tendency, when a different cross section library was used.
The source was designated to ^{60}Co, the main activation source of the spent fuel cask, and ^{18}F having almost the same energy strength to FP-gamma of the spent fuel. ANISN-JR and MCNP5 were used for the calculations with cross section libraries listed in Table 2. Mcplib used for MCNP5 has continuous energy structure, MATXSLIB and VITAMIN-B6 has 42. energy group structure, while DLC23/CASK has 18 groups structure.
Figure 2-1 shows the calculated gamma dose from ^{60}Co and Fig.2-1 shows calculated gamma dose from ^{18}F. The dose rate described in Ref. [10] was added to Fig.2 as substitute for experimental measurements.

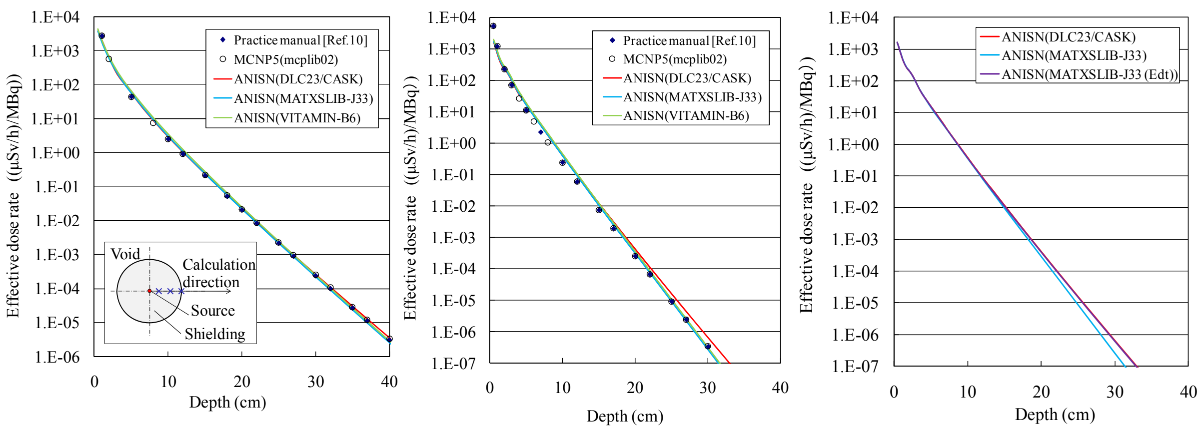

Fig. 2-1. ^{60}Co source   Fig. 2-2. ^{18}F source   Fig. 2-3. ^{18}F source

Fig. 2 Attenuation of gamma dose in iron

In Fig.2-1, all libraries show almost the same gamma dose attenuation. On the other hand, only DLC23/CASK overestimated the measurement shown in Fig.2-2, the model dealing with ^{18}F as a source. This overestimation may be caused by the difference of energy group structure among libraries, as listed in Table 2 below.

Table 2 Energy group structure of libraries

mcplib02 (MCNP5)	Continuous energy
MATXLIB-J33	42 groups
VITAMIN-B6	42 groups
DLC23/CASK	18 groups
MATXLIB-J33 (Edt)	18 groups

The dose calculated by using MATXLIB-J33(Edt) was compared with that using DLC23/CASK, as shown in Fig.2-3, to examine the cause of overestimation of DLC23/CASK. MATXLIB-J33(Edt) is an edited library having 18 energy-group structure as contracted from MATXLIB-J33. The dose calculated by using MATXLIB-J33(Edt) replicated that by using DLC23/CASK. Therefore, the cause of overestimation by using DLC23/CASK was confirmed as being the coarse energy-group structure of DLC23/CASK. In other words, the energy-group structure of libraries has a strong influence of the calculation results. Moreover, a 50% improvement in calculation accuracy can be expected when using MATXLIB-J33 or VITAMIN-B6 instead of DLC23/CASK.

1.2 Examination for neutron

When calculating the neutron doses with a multiply-layered model, it is difficult to comprehend the attenuation tendency in respective materials without the back-scattering effect. Therefore, doses were calculated with a cylinder geometry model composed of 75.9 cm thick fuel and 150 cm thick shielding as shown in Fig.3. Iron was employed because of the remarkable difference of attenuation, attributed to cross section library.

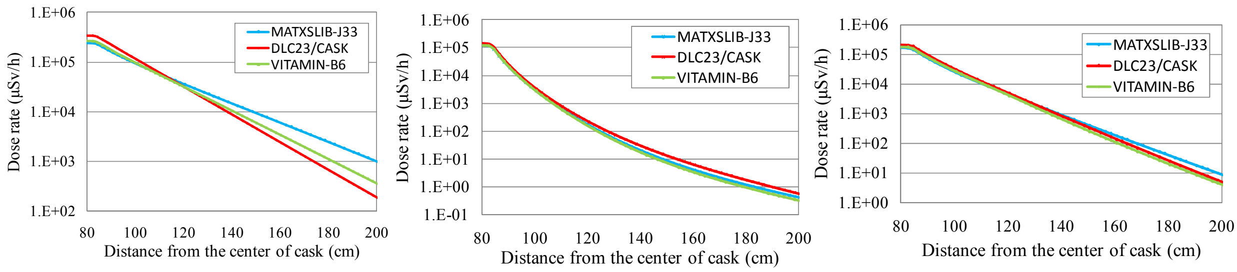

Fig. 3-1. In iron   Fig. 3-2. In resin   Fig. 3-3. In concrete

Fig. 3 Attenuation curve of neutron dose

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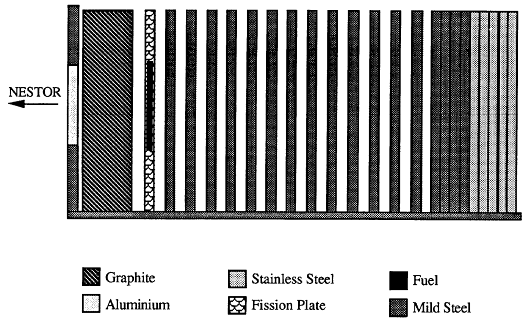
Dose calculations were conducted by using DLC23/CASK, MATXSLIB-J33 and VITAMIN-B6 to examine the attenuation for neutrons in iron, and then compared with the measurement of ASPIS iron benchmark included in SINBAD [11].

Figure 4-1 shows the cross sectional view of this benchmark experiment. The source is a fission converter plate consisting of natural uranium metal plates driven by a thermal flux from the extended graphite reflector of the NESTOR reactor. The energy spectrum of the source is the one of neutrons emitted from the fission of ^{235}U, with the radial dependence being cosine shaped. The iron shield consists of 24 iron plates 180x190x5.0 cm stacked one behind the other with 0.64 cm air gaps between the adjacent plates to allow foils of detector to be loaded. The detectors used were the foils made by ^{32}S, ^{115}In, ^{103}Rh, and ^{197}Au.

Figure 4-2 shows the total cross section of iron and the neutron-reactive cross sections of detectors mentioned above. The Detectors were selected according to the relationship between the resonant energy of iron and the energy sensitivity of the detectors. Table 3 shows working energy range of detectors.

Calculations were conducted by using the DORT code with the cross section data listed in Table 4 [12].



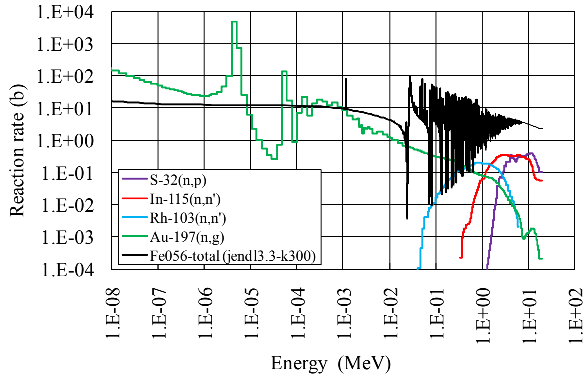


Fig. 4-1. Cross sectional view of the ASPIS iron benchmark

Fig.4-2. Total cross section of iron and neutron reactive cross sections of detectors

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Table 3 Detectors used in ASPIS iron benchmark experiment

Detector	Working energy range
^{32}S(n, γ)	1.6 MeV < E
^{115}In(n, n')	0.4 MeV < E
^{103}Rh(n, n')	40 keV < E
Cd{^{197}Au(n,γ)}	0.55eV-100keV

Table 4 Cross section data used in calculations

	DLC23/CASK	MATXLIB-J33	VITAMIN-B6
^{32}S (n, p)	JENDL/D-99(*1)	JENDL/D-99(*1)	JENDL/D-99(*1)
^{115}In (n, n')	JENDL/D-99(*1)	MATXLIB-J33	VITAMIN-B6
^{103}Rh (n, n')	JENDL/D-99(*1)	JENDL/D-99(*1)	VITAMIN-B6
^{197}Au (n, g), Cd-covd.	(*2)	(*2)	(*2)

(*1) Group constant file edited using JENDL/D-99(JENDL Dosimetry File 99) by NJOY [13,14]
(*2) Calculation report ［11］

Figures 5-1, 5-2, 5-3, and 5-4 show the tendencies of attenuation in iron as detected by ^{32}S, ^{115}In, ^{103}Rh, and ^{197}Au respectively. All attenuation curves of the reaction rate in Fig.5-1 show good agreement with the measurement. Both Fig. 5-2 and Fig. 5-3 show different tendencies of attenuation depending on libraries, for iron is more than 50cm thick. Particularly, when calculating the reaction rate by using DLC23/CASK or VITAMIN-B6 underestimate the measurement detected by ^{103}Rh. However, in the range of 20-40cm, all attenuation curves in Fig.5-2 and Fig. 5-3 show good agreement with the measurement values. In Fig.5-4, the reaction rate conducted by using

DLC23/CASK behaves differently compared to other libraries. This seemed to be caused by exclusion of the self-shielding effect, under the range of resonance energy of iron.

When calculating the model with the thick shieldings of iron with DLC23/CASK or VITAMIN-B6, the measurement values could possibly be underestimated. Conversely, MATXLIB-J33 can be applied to the shielding design regardless of shielding thickness, although the actual value could be overestimated.

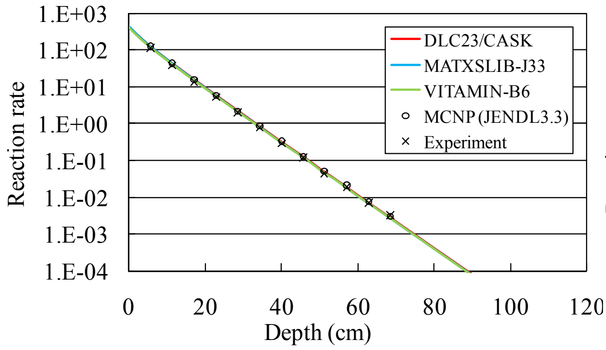

Fig. 5-1. ^{32}S (n, p) reaction

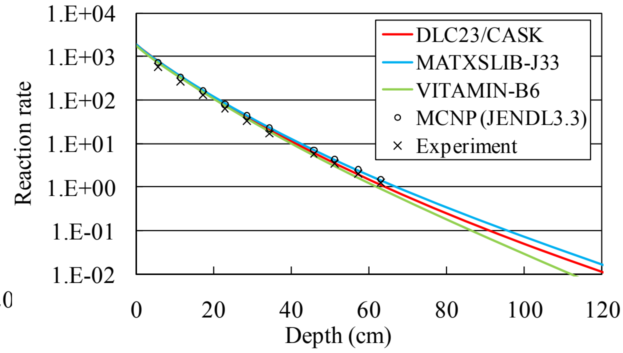

Fig. 5-2. ^{115}In (n, n') reaction

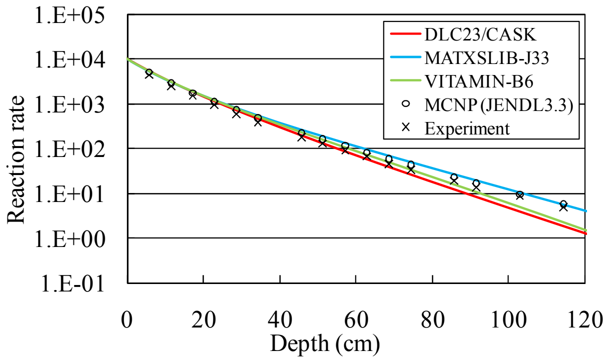

Fig. 5-3. ^{103}Rh (n, n') reaction

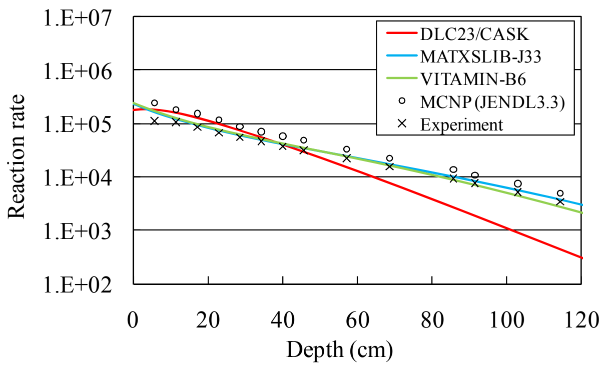

Fig. 5-4. ^{197}Au(n, g) reaction

Fig. 5. Dose attenuation in iron simulating ASPIS iron benchmark experiment

2. EXAMINATION OF CROSS SECTION LIBRARIES FOR SOURCE TERM CALCULATIONS

When evaluating the neutron source strength of casks, calculations are generally conducted by using the ORIGEN code for its easier usage than other codes, such as SRAC, SWAT and MVP-BURN [15,16,17]. BWRU the cross section library based on ENDF/B-IV involved in the ORIGEN code, is generally employed in Japan. However, BWRU was pointed out to underestimate the amount of ^{244}Cm which is main neutron source accounting for about 90% of all neutrons emitted from fuel rod [18].Therefore, calculations with BWRU were conducted and compared with the measurement values especially with focus on ^{244}Cm in order to confirm the applicability to design of a cask. In addition, calculations with ORLIB-J33 based on JENDL3.3 and with GE8x8-4 based on ENDF/B-VI also conducted for the same purpose.

2.1 Description of benchmark and calculation conditions

SF98 in SF-COMPO, a public benchmark measurement database for spent fuel rods, was employed on grounds including the measurement of ^{244}Cm [19]. Fig.6 shows the Burn-up history and sampling points of fuel rod. Sampling points, SF98-4 to SF98-8, were selected for their irradiation spectrum stability, this means upper and lower ends of fuel rod, and the boundary between natural uranium and enriched uranium were excluded as samples. In this case, the average location of these five points is almost at the center of the fuel rod. The initial enrichment of UO_2 is 3.9%, with a burn-up range is 27.2-44.0GWd/tU and average void ratio is 43%.

Calculations were conducted by ORIGEN2.2 or ORIGEN-ARP with a precise burn-up history [20,21]. Void ratio was set to 40% or 70% when using ORLIB-J33, 43%, 50%, or 60% when using GE8x8-4.

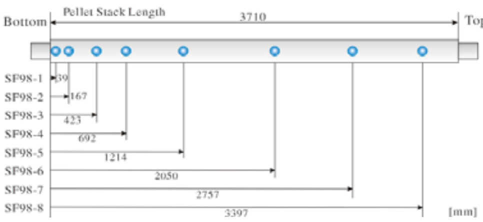

Fig. 6. Sampling points of fuel.

Table 5 Burn-up history

Status	burn-up	cool	burn-up	cool	burn-up	cool	burn-up	cool	burn-up
Days	141	21	257	117	322	9	86	81	368

2.2 Results and discussion

Table 6 shows below lists relative product content of ^{244}Cm calculated as C/E, calculation-value/experiment measured value, and the averages of these values. Average values were calculated as the weighted average.

Table 6 C/E of ^{244}Cm

	ORIGEN2.2			ORIGEN-ARP		
	BWRU	ORLIB-J33 (JENDL3.3)		GE8x8-4 (ENDF/B-VI)		
		BS240J33 (Void ratio = 40)	BS270J33 (Void ratio = 70)	Void ratio = 43	Void ratio = 50	Void ratio = 60
SF98-4	1.40	1.24	1.53	1.29	1.35	1.45
SF98-5	1.23	1.08	1.32	1.10	1.16	1.24
SF98-6	0.87	0.77	0.97	0.82	0.87	0.94
SF98-7	0.78	0.70	0.88	0.74	0.79	0.85
SF98-8	0.56	0.52	0.74	0.67	0.72	0.80
Average	**1.03**	0.92	**1.14**	0.96	**1.02**	**1.09**

Calculation results obtained by using BWRU shows good agreement with measured one from only average value listed in Table 5. In addition, the result was considered to have several percent safety margins compared to the actual values. The densities calculated using ORLIB-J33 with a void ratio of 40% underestimate the measured ones slightly and ORLIB-J33 with a void ratio of 70% overestimates the measured ones as listed in Table 5. BS270J33 is thus considered to be better for the shielding design of fuel casks because the calculated densities with this library have reasonable safety margins compared to the measured value. When calculating by using ORIGEN-ARP, an arbitrary number can be used as the void ratio. The densities calculated using GE8x8-4 with a void ratio of 43%, average void ratio of this fuel rod, underestimate the measured one. Calculations were also performed with 50% and 60% void ratio as reference and lead 1.02, 1.09 as results. The void ratio is recommended to be set more than 50% when using GE8x8-4 for shielding designs to avoid underestimating.

CONCLUSIONS

１） MATXSLIB-J33 was found to be the best library among those due to its better agreement with measured values. When calculating with DLC23/CASK, gamma dose could be overestimated and the neutron dose underestimated. When calculating with VITAMIN-B6, the neutron dose could also be underestimated.

２） When calculating the neutron strength of a cask by using ORIGEN2.2 with BWRU, calculation results were considered to have several percent safety margins compared to measured values. On the other hand, when using ORIGEN2.2 with ORLIB-J33 or ORIGEN-ARP with GE8x8-4, the results could be underestimated without appropriate setting of the void ratio.

REFERENCES

[1] Kinsey R. 1979, Data format and procedures for the evaluated nuclear data file,ENDF,BNL-NCS-50496 2nd Ed.
[2] ORNL-RSIC, "CASK-40 Group Coupled Neutrons and Gamma-ray Cross section Data," DLC-23, Sep. 1973
[3] K.Shibata, et al., , J. Nucl. Sci. Technol., 39, p.1125 (2002)
[4] K. Kosako, N. Yamano, T. Fukahori, K. Shibata and A. Hasegawa, "The Libraries FSXLIB and MATXSLIB based on JENDL-3.3," JAERI-Data/Code 2003-011 (2003)
[5] (Comp. and Ed.) P.F. Rose: "ENDF-201, ENDF/B-VI Summary Documentation," BNL-NCS-17541, 4th Edition (1991).
[6] J. E. White, et al., "VITAMIN-B6: A Fine-Group Cross section Library Based on ENDF/B-VI for Radiation Transport Applications, Gatlinburg, Tennessee, p.733-736 (1994).
[7] J. Katakura, et al., "A Set of ORIGEN2 Cross section Library Based on JENDL-3.3 Library: ORLIBJ33," JAERI-Data/Code 2004-015, Nov. 2004.
[8] W. W. Engle, Jr., "ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693 (March 1967)
[9] X-5 Monte Carlo Team at the Los Alamos National Laboratory, "MCNP -A General Monte Carlo N-Particle Transport Code, Version 5," LA-UR-1987, (2003).
[10] Nuclear Safety Technlogy Center "Practice Manual for shielding calculation 2000"
[11] J. Butler, et al.,SINBAD ABSTRACT NEA-1517/34 Winfrith Iron Benchmark Experiment (ASPIS)
[12] J. O. Johnson, Ed., DORT section of "A User's Manual for MASH 1.0 – A Monte Carlo Adjoint Shielding Code System," ORNL/TM-11778 (1992).
[13] R. E. MacFarlane and D. W. Muir, "The NJOY Nuclear Data Processing System Version 91," LA-12740-M (October 1994).
[14] K.Kobayashi, et al., "JENDL Dosimetry File 99 (JENDL/D-99)," JAERI 1344 (2002).
[15] Okumura K. ,Kaneko K,Dobashi K, "SRAC95",JAERI-Data/Code 96-015 (1996).
[16] K. Suyama, et al., "Integrated Burn-up Calculation Code System SWAT," JAERI-Data/Code 97-047, Nov. 1997.
[17] Okumura K. , et al., "Validation of a Continuous-Energy Monte Calro Burn-up Code MVP-BURN and Its Application to Analysis of Post Irradiation Experiment", *J.Nucl.Sci. Technol.*, 37,128(2000).
[18] Okumura K. , et al., Nuclear data News No.74,P39-55(2003)
[19] Science and Technology Agency of Japan, Post Irradiation Examination for the Spent Fuel Samples
[20] A.G.Croff, Nucl. Technol., 62,p335(1983).
[21] I. C. Gauld, et al., "ORIGEN-ARP: Automatic Rapid Processing for Apent Fuel Depletion, Decay, and Source Term Analysis," ORNL/TM-2005/39 (2006)