ORLIBJ32 : THE SET OF NEW LIBRARIES OF ORIGEN2 CODE BASED ON JENDL-3.2

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ABSTRACT

A set of new libraries for ORIGEN2 code, “ORLIBJ32,” based on Japanese evaluated nuclear data library was developed. The libraries contain not only one grouped cross section data but also variable actinide cross section and decay and fission yield data. Main objectives of the libraries are to calculate isotopic compositions averaged over whole fuel assemblies for the case of LWR and isotopic compositions averaged over a whole core for the case of FBR. Target fuel assemblies are PWR 17×17 type and BWR 8×8 or 9×9 type. Target core and blanket types of FBR are chosen from several considered specification. Evaluation of the LWR libraries was conducted by the analysis of latest post irradiation examinations carried out in Japan Atomic Energy Research Institute. The evaluation showed improved results of many isotopes. Evaluation of the FBR libraries was carried out by the comparison between new and old libraries of FBR. The calculated weights of several isotopes presented large differences. This comparison presents that the differences of calculated results depend on the neutron spectrum of the

target systems.

1. INTRODUCTION

ORIGEN2[1] is one of the most widely used burnup codes for many purposes, such as analysis of transmutation of radioactive waste. It is because of its simple and user friendly input data, rich contents of output file, and an advantage of data libraries prepared for various reactor systems.

However, because ORIGEN2 was mainly developed from the end of 1960's to early 1980's, the libraries in ORIGEN2 should be improved. One issue is that the libraries are based on old design of reactors and the other is that the used nuclear data files are old such as ENDF/B-IV[2], ENDF/B-V[3], and LENDL[4]. Also because the burnup value of spent fuel has been increased today, suitable libraries for high burnup and high enrichment fuel for ORIGEN2 code is requested. To overcome these problems, Japanese Nuclear Data Committee (JNDC) launched the project to make libraries of ORIGEN2 code based on the current reactor design using the newest nuclear data file JENDL-3.2[5] developed in Japan.

In this study, the development the set of new ORIGEN2 libraries “ORLIBJ32"[6] based on JENDL-3.2 is shown. In section 2, the objectives of the libraries are shown. Method of development of the LWR and FBR libraries are presented in section 3. To make cross section libraries for LWR, integrated burnup code system SWAT[7] was used. To make cross section libraries for FBR, core calculation system used in Japan Nuclear Cycle Development Institute (JNC) was modified and used. Also, in section 3, the update of decay and fission yield data was described. In section 4, examples of the evaluation of the ORLIBJ32 are shown. In the evaluation of LWR libraries, latest post irradiation examination (PIE) carried out in Japan Atomic Energy Research Institute(JAERI) was used. Since no available PIE data exists, the calculated weights of isotopes are compared for the case of FBR libraries. Section 5 is conclusion.

2. OBJECTIVE LIBRARIES

Four kinds of libraries are prepared in ORIGEN2 as shown in **Table I**. “The one-grouped effective cross section libraries” are data storage files of one-grouped effective cross sections of various reactor systems. “The variable actinide cross sections” are subroutines written in FORTRAN for each reactor system. To use the subroutines, a source program of ORIGEN2 should be modified, recompiled and linked. The subroutines contain the one-grouped effective cross section data of the actinide against several burnup values. “The libraries of decay constant and fission yield data” are the database of decay constants, branching ratios of radioactive decay, recoverable energy of decay, and direct fission yield data, etc. “The photon spectrum data library” has 18 group photon spectrum data for various radio active isotopes.

Table I List of Data Library in ORIGEN2 Code

Name of Libraries	Make New Library (Yes or No)
One grouped cross section data	Yes
Variable actinide cross section data	Yes
Decay constants and fission yield data	Yes
Photon spectrum data	No

In this project, all the data libraries are compiled for the current reactor systems based on JENDL-3.2 except the photon spectrum data library. The library of decay constants and fission yield data is compiled from the JNDC fission products library 2nd version. It is because these data are not contained in JENDL-3.2.

The specifications for LWR libraries are founded on current fuel design parameters. In **Table II** and **Table III**, main parameters of each library are shown. For PWR, a 17×17 fuel assembly is a target, and for BWR, 8×8 (step I and II) and 9×9 (step III) fuel assemblies. For FBR libraries, the parameters are chosen from the requests of analysts of fast reactors since no fixed specifications of FBR exist, especially for a commercial fast reactor. The libraries for FBR were made for each type of core (JOYO, MONJU, 600 MW prototype reactor, 1300 MW commercial reactor, or Pu burner) and for each type of fuel (MOX, metal, or nitride fuel) and blanket (inner or outer).

Table II List of LWR libraries

Libraries	Fuel Assembly	Void Ratio(%)	U-235 Enrichment(%)	Maximum Burnup (GWd/t)
BS100J32	BWR STEP-1	0	3.0	40
BS140J32	BWR STEP-1	40	3.0	40
BS170J32	BWR STEP-1	70	3.0	40
BS200J32	BWR STEP-2	0	3.8	50
BS240J32	BWR STEP-2	40	3.8	50
BS270J32	BWR STEP-2	70	3.8	50
BS300J32	BWR STEP-3	0	4.0	60
BS340J32	BWR STEP-3	40	4.0	60
BS370J32	BWR STEP-3	70	4.0	60
PWR34J32	PWR 17×17	-	3.4	60
PWR41J32	PWR 17×17	-	4.1	60
PWR47J32	PWR 17×17	-	4.7	60

Maximum burnup means that the upper burnup value for interpolation of variable actinide cross section data.

Table III List of FBR Libraries

Reactor Type (fuel)	Core		Blanket	
	Inner	Outer	Axial	Radial
JOYO(MOX)	JOYOM1CO		JOYOM1AX	JOYOM1RD
MONJU(MOX)	MONJMXIC	MONJMXOC	MONJMXAX	MONJMXRD
600 MWe(MOX)	600MMXIC	600MMXOC	600MMXAX	600MMXRD
600MWe(Metal)	600MMTIC	600MMTOC	600MMTAX	600MMTRD
600MWe(Nitride)	600MNIIC	600MNIOC	600MNIAX	600MNTRD
600MWe(recycled Pu)	600MRPIC	600MRPOC	600MRPAX	600MRPRD
1300MWe(MOX)	1300MXIC	1300MXOC	1300MXAX	1300MXRD
Pu Burner	PUBRMXIC	PUBRMXOC	-	-

3. METHOD

3.1 CROSS SECTION

A main object to make LWR libraries is to evaluate assembly-averaged isotopic composition. For this object, we adopt a single pin cell model that explains the target assembly to evaluate an effective cross section and a neutron spectrum. The single pin model has the same H/U ratio with the target assembly, and the dancoff correction factor is taken as the averaged value of each fuel pin in the assembly. **Figure 1** is schematic figure describing the method to make the single pin cell model.

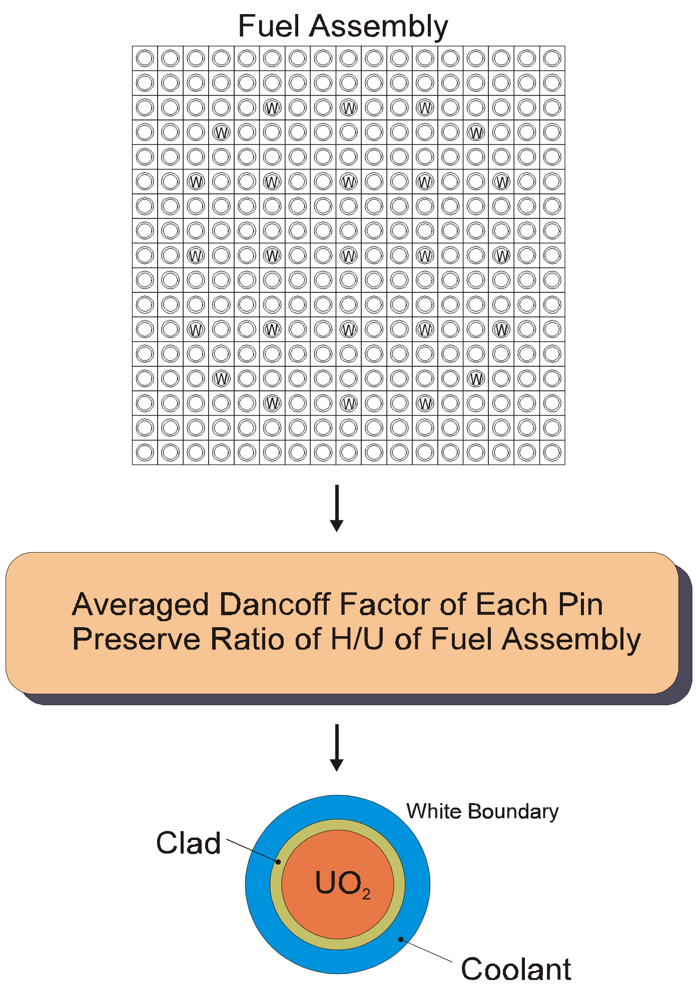

Figure 1. Schematic Figure of Single Pin Cell Model Used to Make LWR Libraries

To make the LWR libraries, the integrated burnup code system SWAT was used as shown in **Figure 2**. Using SWAT, we performed burnup calculation adopting the single pin cell model based on specified burnup parameters such as power history. By this calculation, SWAT generates the one group cross section data file that depends on the value of burnup. The one group cross section data file prepared by SWAT has same format of the libraries of ORIGEN2 code. And SWAT2ORI2 package process the one group cross section data file to make a variable actinide cross section subroutine to be linked to a source program of ORIGEN2.

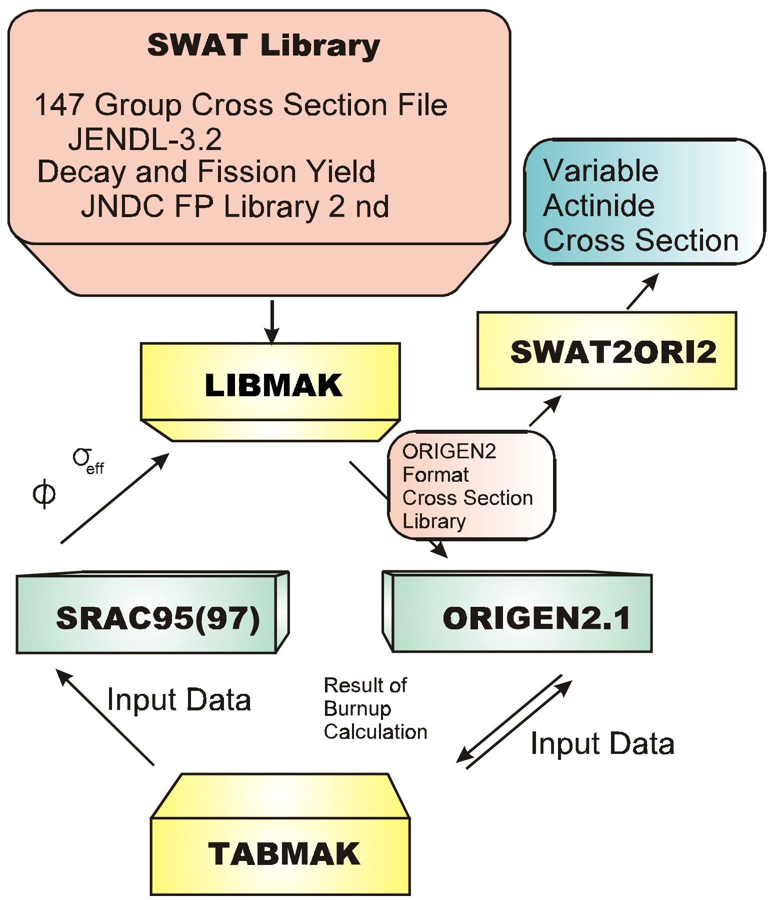

Figure 2. Flow of Calculation using SWAT to make new ORIGEN2 libraries

The path to make the new libraries for FBR differs from the case of LWR. To compile FBR libraries, a new system[8] is developed based on the standard core calculation system used in Japan Nuclear Cycle

Development Institute (JNC).

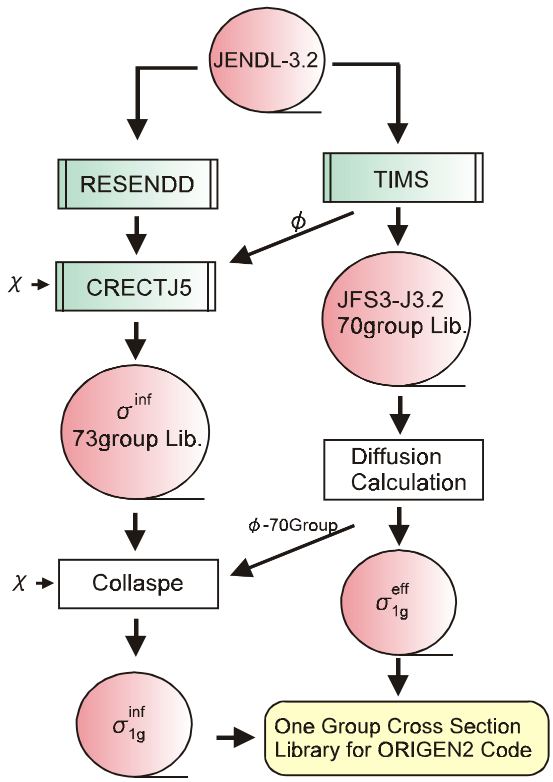

Figure 3. Flow of making FBR libraries

The new system is shown in **Figure 3**. As this figure shows, an infinite diluted cross section library of 73 groups is compiled with CRECTJ5[9] using a neutron spectrum which is prepared by connecting a standard neutron spectrum and a prompt fission neutron spectrum of Pu-239. The standard spectrum is 70 groups and used for making JSF3-J3.2[10]. Also, diffusion calculation is carried out for target reactors with JFS3-J3.2. We obtain the neutron spectrum and effective cross section of 70 groups by that calculation, and effective one group cross section data is generated by collapsing the 70 groups data. Then, data in the 73 groups library are collapsed to infinite diluted one grouped data with a 73 group neutron spectrum prepared by connecting the neutron spectrum from that diffusion calculation and the prompt fission neutron spectrum of Pu-239. Finally, the infinite diluted cross sections are substituted with the effective cross section data. This system will be distributed for users to make a new library for satisfying each user's purpose.

Since for the case of FBR, dependencies of one grouped cross section on the burnup are small, constant values of cross sections were set in the subroutine of variable actinide cross section.

3.2 DECAY CONSTANT AND FISSION YIELD

The decay constant and fission yield data libraries are also compiled based on JNDC FP Library second version[11] (JNDC-V2) for each of LWR and FBR systems. The FP generation path in JNDC FP Library differs from the path in ORIGEN2 libraries. The path data in JNDC FP Library were modified to fit the data in ORIGEN2 libraries. This means that these new libraries can be used without any modification of a source program of ORIGEN2 except the subroutine of variable actinide cross section.

The fission yield in JNDC-V2 is compiled as the values against the energy of thermal, fission or 14 MeV neutron. In the ORLIBJ32, fission yield data are evaluated by weighting the data in the JNDC-V2 with fission ratio in the thermal energy region and fast energy region.

The decay heat calculation using these libraries is able to reproduce the recommended decay heat values by the Atomic Energy Society of Japan within 5 %[12].

4. EVALUATION

4.1 LWR LIBRARIES

To evaluate the ORLIBJ32, new Post Irradiation Examinations (PIE) data were analyzed. For the evaluation, irradiation parameters of fuel should be the same with the parameters used when libraries were made. We have suitable PIE data sets[13,14] of PWR from the view point of neutronics calculation. JAERI has PIE activity to develop the method of criticality control of spent nuclear fuel from LWR. In that PIE, data set named SF95 (contains five samples' data, 14.7 ~38.1 GWd/t) and SF97 (contains five samples' data, 31.3 ~ 48.2 GWd/t) were taken from 17×17 type fuel assembly irradiated in Japanese commercial PWR. SF97 is the data set from heigh burnup PWR fuel assembly. These fuel assemblies have same initial enrichment of U-235, and the geometrical parameters are completely the same with the parameters used to make PWR libraries (PWR41J32) of ORLIBJ32. The neutron spectrum around SF95 and SF97 samples was not disturbed since the position of irradiation is away from the control rods. In **Table IV and Table V**, ratios of calculated results by ORIGEN2.1 with ORLIBJ32 to experimental results are shown.

Table IV. Ratios of calculated results to experimental results (C/E values) of SF95 Sample by ORIGEN2.1 with ORLIBJ32(PWR41J32)

Sample	SF95-1-1	SF95-1-2	SF95-1-3	SF95-1-4	SF95-1-5	Averaged C/E	Standard Deviation
GWd/t	14.7	25.2	36.7	38.1	31.4		
U-234	1.02	0.92	1.17	1.15	0.84	1.02	0.14
U-235	1.00	1.00	0.97	1.00	1.01	1.00	0.01
U-236	0.96	0.94	0.96	0.96	0.95	0.95	0.01
U-238	1.00	1.00	1.00	1.00	1.00	1.00	0.00
Pu-238	0.84	0.77	0.88	0.93	0.92	0.87	0.06
Pu-239	1.02	0.97	0.98	1.01	1.04	1.01	0.03
Pu-240	1.01	0.99	1.02	1.04	1.05	1.02	0.03
Pu-241	1.01	0.95	0.97	1.02	1.05	1.00	0.04
Pu-242	0.98	0.93	0.98	1.00	1.02	0.98	0.03
Am-241	0.82	1.12	1.13	1.63	1.17	1.17	0.29
Am-242m	0.63	0.60	0.61	0.68	0.73	0.65	0.05
Am-243	0.87	0.87	0.95	1.02	1.02	0.95	0.07
Cm-242	0.76	0.67	0.63	0.57	0.87	0.70	0.12
Cm-243	0.58	0.54	0.65	0.67	0.64	0.62	0.05
Cm-244	0.77	0.67	0.82	0.88	0.95	0.82	0.11
Cm-245	0.87	0.68	0.85	0.90	1.01	0.86	0.12
Cm-246	0.42	0.50	0.83	0.89	1.50	0.83	0.43
Cs-137	1.00	0.99	1.00	1.00	1.00	1.00	0.01
Cs-134	0.93	0.92	0.99	1.01	1.00	0.97	0.04
Eu-154	0.88	0.77	0.77	0.77	0.82	0.80	0.05
Ce-144	0.98	0.99	0.97	1.06	0.99	1.00	0.04
Nd-142	0.83	0.98	0.92	0.95	1.02	0.94	0.07
Nd-143	0.97	0.96	0.94	0.95	0.96	0.95	0.01
Nd-144	1.02	1.03	1.06	1.01	1.03	1.03	0.02
Nd-145	1.00	1.01	1.01	1.01	1.01	1.01	0.00
Nd-146	1.01	1.01	1.02	1.02	1.02	1.01	0.00
Nd-148	1.01	1.01	1.02	1.02	1.02	1.02	0.00
Nd-150	0.98	1.00	1.00	1.00	1.01	1.00	0.01

Table V. Ratios of calculated results to experimental results (C/E values) of SF97 Sample by ORIGEN2.1 with ORLIBJ32(PWR41J32)

Sample	SF97-1-2	SF97-1-3	SF97-1-4	SF97-1-5	SF97-1-6	Averaged	Standard
GWd/t	31.3	43.0	48.0	48.2	41.6	C/E	Deviation
U-234	1.01	0.97	0.96	0.96	0.98	0.98	0.02
U-235	0.98	0.96	0.98	1.01	1.03	0.99	0.03
U-236	0.96	0.95	0.95	0.95	0.95	0.95	0.00
U-238	1.00	1.00	1.00	1.00	1.00	1.00	0.00
Np-237	0.94	0.98	0.98	0.97	0.99	0.97	0.02
Pu-238	0.84	0.84	0.86	0.87	0.92	0.87	0.03
Pu-239	1.00	0.99	1.03	1.04	1.09	1.03	0.04
Pu-240	1.04	1.05	1.05	1.07	1.08	1.06	0.02
Pu-241	0.98	0.99	1.03	1.04	1.08	1.02	0.04
Pu-242	0.99	1.00	1.00	1.00	1.01	1.00	0.01
Am-241	1.25	1.24	1.17	1.16	1.40	1.24	0.10
Am-242m	0.72	0.68	0.67	0.69	0.81	0.71	0.06
Am-243	0.91	0.95	0.97	0.98	1.03	0.97	0.04
Cm-242	1.01	1.10	1.19	1.29	1.17	1.15	0.11
Cm-243	0.62	0.66	0.70	0.71	0.74	0.69	0.05
Cm-244	0.77	0.81	0.85	0.87	0.94	0.85	0.06
Cm-245	0.76	0.81	0.88	0.92	1.06	0.89	0.12
Cm-246	0.75	0.79	0.81	0.82	0.92	0.82	0.06
Cm-247	0.66	0.73	0.79	0.84	0.87	0.78	0.09
Cs-137	1.00	1.00	1.01	1.01	1.00	1.00	0.01
Cs-134	0.91	0.97	1.03	1.04	1.02	0.99	0.05
Eu-154	0.76	0.77	0.79	0.81	0.84	0.79	0.03
Ce-144	0.91	0.99	1.08	1.09	0.97	1.01	0.08
Nd-143	0.97	0.95	0.96	0.96	0.97	0.97	0.01
Nd-144	1.06	1.04	1.01	1.00	1.01	1.02	0.02
Nd-145	1.02	1.02	1.02	1.02	1.01	1.02	0.00
Nd-146	1.01	1.01	1.02	1.01	1.01	1.01	0.00
Nd-148	1.03	1.03	1.03	1.03	1.03	1.03	0.00
Nd-150	1.02	1.02	1.03	1.03	1.03	1.03	0.01
Sm-147	1.01	0.97	0.94	0.94	0.96	0.97	0.03
Sm-148	1.11	1.12	1.12	1.12	1.16	1.13	0.02
Sm-149	0.76	0.82	0.94	0.97	0.86	0.87	0.09
Sm-150	1.03	1.03	1.04	1.03	1.05	1.04	0.01
Sm-151	0.86	0.89	0.94	0.96	1.01	0.93	0.06
Sm-152	1.24	1.26	1.25	1.23	1.21	1.24	0.02
Sm-154	1.00	1.01	1.01	1.01	1.03	1.01	0.01

In **Table VI** and **VII**, averaged ratios of calculated results to experimental results (C/E values) are shown. In these tables, averaged C/E values of U-235, Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242 of SF95 and SF97 samples calculated with PWR41J32(PWR 17×17 assembly: U-235 enrichment is 4.1 weight %) of ORLIBJ32 or PWR-UE, PWR-US and PWR-U are shown. As **Table VI** and **Table VII** show, calculated results using PWR41J32 except PU-238 is improved from original libraries. For example, averaged C/E value of U-235 of SF95 by PWR41J32 is 1.00, however averaged C/E values of SF95 by PWR-UE is 1.05. Pu-242 also improved. However, results of Pu-238 become worse. The underestimation of Pu-238 was reported in previous analysis[13] by SWAT. This is a problem of JENDL-3.2. In these results, large differences of PWR-US and PWR-U come from the difference of objective fuel assemblies of the libraries.

For the case of SF97, the higher burnup fuel samples, the improvement by ORLIBJ32 is more clear. Results by PWR41J32 shows good agreement with experimental results, however the results by PWR-UE show underestimations of U-235 more than 10% and overestimation of Pu-240 more than 10%. In **Figures 4**, **5**, **6** and **7,** comparison between PWR41J32, PWR-UE and PWR-US are shown. These figures also present the same improvement of PWR41J32 from PWR-UE as the **Tables VI** and **VII**, and show the tendency of overestimation or underestimation by PWR-UE library in high burnup region (> 40 GWd/t).

Table VI. Averaged C/E Values for 5 Samples from SF95 PIE data (Uranium and Plutonium)

Isotopes	Libraries			
	PWR41J32	PWR-US	PWR-UE	PWR-U
U-235	1.00	0.88	0.95	0.91
Pu-238	0.87	0.83	1.02	0.80
Pu-239	1.01	0.85	0.98	0.82
Pu-240	1.02	0.95	0.99	1.02
Pu-241	1.00	0.78	0.95	0.77
Pu-242	0.98	0.83	0.92	0.81

Table VII. Averaged C/E Values for 5 Samples from SF97 PIE data (Uranium and Plutonium)

Isotopes	Libraries			
	PWR41J32	PWR-US	PWR-UE	PWR-U
U-235	0.99	0.76	0.89	0.81
Pu-238	0.87	0.92	1.09	0.85
Pu-239	1.03	0.91	1.07	0.83
Pu-240	1.06	1.05	1.11	1.03
Pu-241	1.02	0.81	0.92	0.84
Pu-242	1.00	0.84	0.87	0.88

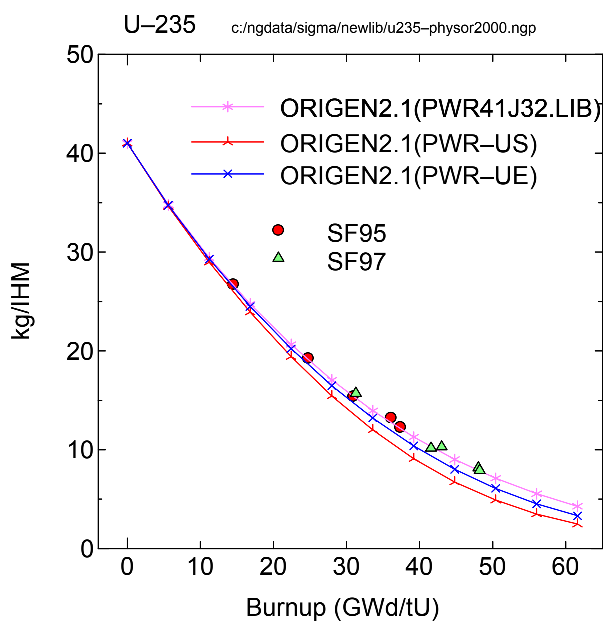

Figure 4. Comparison between calculation results using three libraries: U-235

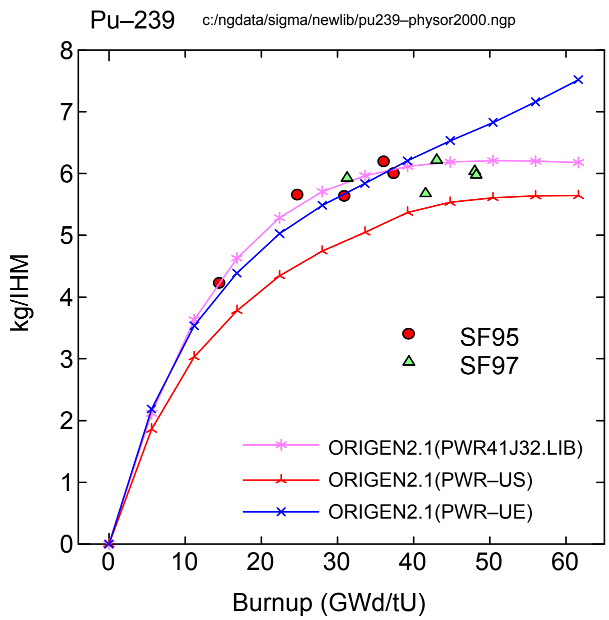

Figure 5. Comparison between calculation results using three libraries: Pu-239

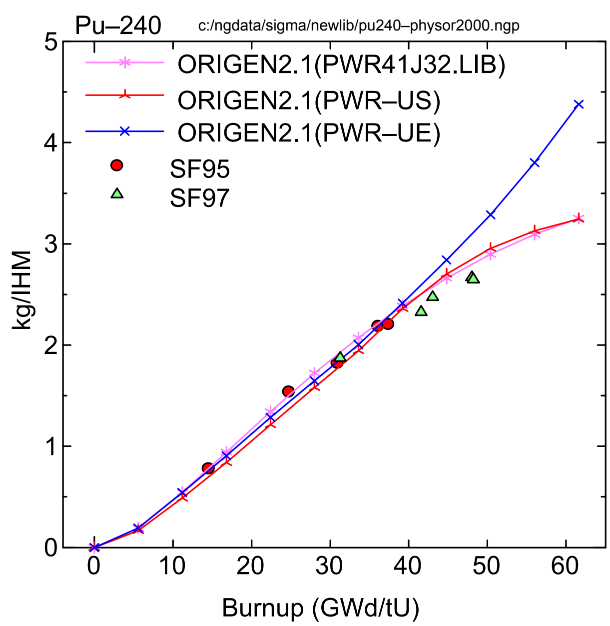

Figure 6. Comparison between calculation results using three libraries: Pu-240

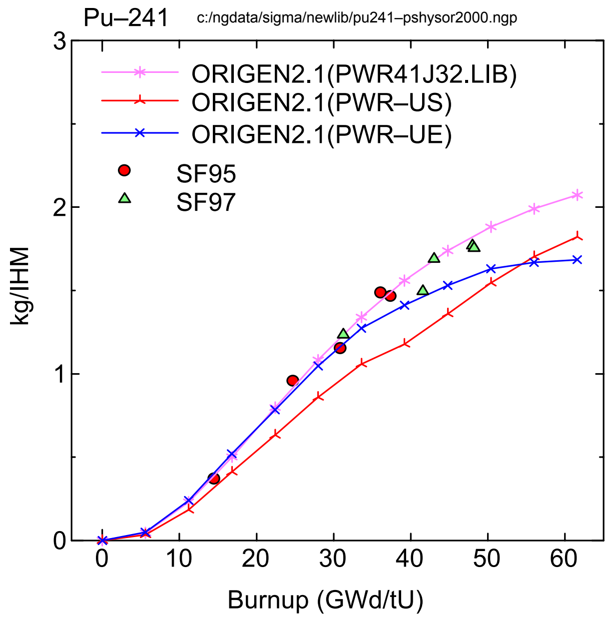

Figure 7. Comparison between calculation results using three libraries: Pu-241

In **Table VIII**, the averaged C/E values of SF95 for FP are shown. As this table shows that Eu-154 shows large differences. However, the other isotopes show good agreement with experimental results. For FP, Samarium isotope is one of the most interesting from the view point of reactivity evaluation.

Table VIII. Averaged C/E Values for 5 Samples from SF95 PIE data (Fission Products)

Isotopes	Libraries			
	PWR41J32	PWR-US	PWR-UE	PWR-U
Cs-134	0.97	0.91	0.99	0.89
CS-137	1.00	0.97	0.97	0.97
Nd-143	0.95	0.96	0.97	0.95
Eu-154	0.80	1.25	1.37	1.49

The PIE data of Samarium isotopes are taken in SF97 data set. In **Table IX**, the averaged C/E values of

SF97 for FP are shown including Samarium results. **Table IX** shows the calculated results of FP by PWR41J32 except Sm-149 are improved from original libraries.

The reason of the difference of Sm-149 should be resolved. One reason of the difference is Sm-149 is affected by the power history end of the cycle because of the generation chain. The other reason is the cross section data during burnup. In the ORIGEN2 calculation, cross section data of FP are fixed.

Table IX. Averaged C/E Values for 5 Samples from SF97 PIE data (Fission Products)

Isotopes	Libraries			
	PWR41J32	PWR-US	PWR-UE	PWR-U
Cs-134	0.99	0.98	1.04	0.92
CS-137	1.00	0.98	0.97	0.98
Nd-143	0.97	0.94	0.98	0.93
Eu-154	0.79	1.51	1.61	1.70
Sm-147	0.97	0.78	0.75	0.80
Sm-148	1.13	1.24	1.27	1.22
Sm-149	0.87	0.95	1.08	0.92
Sm-150	1.04	1.12	1.15	1.12
Sm-151	0.93	1.12	1.35	1.07
Sm-152	1.24	1.30	1.27	1.33
Sm-154	1.01	1.02	1.06	1.04

For the case of BWR libraries, we compared the calculated results using ORLIBJ32 (BS240J32 or BS270J32) with the results by assembly calculation code TGBLA[15]. TGBLA is neutronics code used in the BWR fuel design and analysis. In **Table X**, ratios of the ORIGEN2 results with ORLIBJ32 to TGBLA results are shown. As this table shows, the deviations of the results by ORLIBJ32 from TGBLA are less than 10%. These deviations are enough to satisfy the users of ORIGEN2 code.

Table X. Comparison of Isotopic Composition between ORIGEN2 and Assembly Calculation (BWR STEP-2 Fuel: 40 GWd/t)

Isotopes	ORIGEN2/TGBLA	
	Void Ratio = 40 %*	Void Ratio = 70 %**
U-235	0.94	0.96
Pu-239	1.04	1.09
Pu-240	0.94	0.92
Pu-241	1.03	1.08
Pu-242	1.01	1.05

* Using BS240J32 Library

** Using BS270J32 Library

4.1 FBR LIBRARIES[8]

Since no available PIE data for FBR exist, calculated amounts of generation or depletion of isotopes are compared between new and old libraries. In **Table XI**, the change of amounts (weight after irradiation - weight before irradiation) of major isotope was compared. In this case, FBR libraries for MOX fuel in ORLIBJ32 were exchanged from the view point of the types of cores. And in **Table XII**, change of amounts (weight after irradiation - weight before irradiation) of major isotopes was compared. In this case, FBR libraries for 600 MWe FBR in ORLIBJ32 were exchanged from the view point of the types of fuels.

This analysis revealed that the differences of results are large between using original library and ORLIBJ32. For example, change of amount of Pu-240 by ORLIBJ32 are 4 times smaller than the results by original library. These differences come mainly from a change of a neutron spectrum to collapse multi group cross section to one group data since all of the FBR libraries are compiled by collapsing common 73 groups data. This means that we must pay attention to the neutron spectrum when we calculate the isotopic composition of spent fuel from FBR.

The example of uses of this new library is shown in the reference[16]. In that case, decay heat of spent nuclear fuel from JOYO was analyzed by ORIGEN2 with ORLIBJ32, and about 10% differences from experimental data were shown.

Table XI. Comparison of change of amount by calculation between new and old libraries (FBR)[8]:the case of change of core - Burnup: 86,000 Mwd/t
Initial fuel Pu Vector: Pu-238/Pu-239/Pu-240/Pu-241/Pu-242 = 3/53/25/12/7
Old library is LMFBR: Advanced oxide, LWR-Pu/U/U/U (NLB: 311 - 319)

Isotope	Types of Cores* of Libraries in ORLIBJ32				
	JOYO	MONJU	600MWe	1300MWe	Pu Burner
U-235	-12.9**	3.1	4.7	4.7	3.3
Pu-328	-5.2	-1.3	-1.2	-1.3	-1.8
Pu-239	235.4	-23.8	-27.7	-38.1	-19.2
Pu-240	-444.1	65.6	107.5	107.5	105.4
Pu-241	-1.5	-4.6	-4.7	-4.6	-7.1
Pu-242	100.0	-14.3	-23.8	-23.8	-21.4
Am-241	6.7	-0.2	-0.9	-0.9	0.0
Cm-244	-15.2	90.2	103.7	103.4	98.2

*: Inner Core except JOYO

** : (Result with ORLIBJ32 - Result with Original Lib.)/ Result of Original Lib × 100

Table XII. Comparison of change of amount of by calculation between new (600MWe) and old libraries (FBR)[8] : the case of change of type of fuel - Burnup: 86,000 Mwd/t
Initial fuel Pu Vector: Pu-238/Pu-239/Pu-240/Pu-241/Pu-242 = 3/53/25/12/7
Old library is LMFBR: Advanced oxide, LWR-Pu/U/U/U (NLB: 311 - 319)

Isotope	Type of Fuel (Reactor is 600 MWe FBR) of Libraries in ORLIBJ32			
	MOX	METAL	NITRIDE	Pu-Recycle
U-235	4.7*	-1.2	0.4	4.5
Pu-328	-1.2	-0.9	-1.7	-1.4
Pu-239	-27.7	176.9	85.8	-33.1
Pu-240	107.5	-243.0	-101.1	104.3
Pu-241	-4.7	2.4	-1.2	-4.6
Pu-242	-23.8	11.9	7.1	-26.2
Am-241	-0.9	1.9	1.0	-0.9
Cm-244	103.7	31.3	58.8	100.8

* : (Result with ORLIBJ32 - Result with Original Lib.)/ Result of Original Lib × 100

CONCLUSION

The set of new libraries for ORIGEN2 code “ORLIBJ32" were developed based on JENDL-3.2 using the latest core parameters. The targets are the 17×17 fuel assemblies for PWR, and Step I, II or III assembly for BWR. For FBR libraries, several types of cores and fuels were taken for targets, and libraries for not only cores but also blanket regions were developed. To make new libraries for LWR, the integrated burnup code system SWAT was used adopting the single pin cell model. For making FBR libraries, the new system was developed based on core calculation system used in JNC.

The analysis of PIE data from PWR fuel assemblies was carried out and the calculated results showed good agreement with the experimental data, and improvement from original libraries are shown. For FBR, the comparison of the calculated data revealed the large difference between those using the new and old data libraries.

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