日本語題目
JENDL-3.2に基づくORIGEN2用ライブラリの作成

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Development of Libraries for ORIGEN2 Code Based on JENDL-3.2*

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The working Group of JNDC "Nuclide Generation Evaluation" has launched a project to make libraries for ORIGEN2 code based on the latest nuclear data library "JENDL-3.2" for current design of LWR and FBR fuels. Many of these libraries are under validation.

1. INTRODUCTION

ORIGEN2 code[1] is one of the most widely used burnup codes for many purposes, such as analysis of transmutation of radioactive waste, evaluation of the source of radiation. The advantage of using ORIGEN2 is that it has a variety of data libraries for various reactor systems.

However, the libraries in ORIGEN2 have at least two problems. One is that the libraries are based on obsolete design of reactors and the other is that the used nuclear data libraries are old such as ENDF/B-IV, ENDF/B-V, and LENDL for development, since ORIGEN2 code had been mainly developed from the end of 1960's to early 1980's. This means that ORIGEN2 code has no adequate library based on the latest nuclear data libraries for the current reactors.

To overcome these problems, JNDC launched the project to make libraries of ORIGEN2 code based on the current reactor design using JENDL-3.2[2]. The aim of making these libraries is to evaluate assembly averaged isotopic composition as correctly as possible.

This paper describes the background and method developing the new libraries of ORIGEN2 code based on JENDL-3.2.

2. OBJECTIVE

Four kinds of libraries are prepared in ORIGEN2 as shown in **Table 1**. "One-grouped cross

*This study uses the part of the work carried out by the Japan Atomic Energy Research Institute (JAERI) under the entrustment by the Science and Technology Agency (STA) of JAPAN.

section libraries" are data storage of one-grouped cross sections for various reactor systems. For FP and activation products, the one-grouped cross sections of (n, γ), (n,2n), (n,p) and (n, α) are contained in these libraries. Also, for actinides, those of (n, γ), (n,2n), (n,3n), (n,f) are preserved in them. "Subroutines of variable actinide cross section" are source programs written in FORTRAN for each reactor system. They contain the one-grouped cross sections depending on burnup. A user can take into account the effect of change of a neutron spectrum during burnup using one of these subroutines. "Libraries of decay and fission yield" are database of decay constants, branching ratios of radioactive decay, recoverable energy of decay, and direct fission yields, etc. "A photon spectrum library" has 18 grouped photon spectrum data for various radio active isotopes.

In this project, all of the data libraries are compiled for the current reactor systems based on JENDL-3.2 library except for the photon spectrum data library. The library of decay and fission yield, it is compiled from JNDC fission products library 2nd version (JNDCFPV2)[3].

Table 1 Libraries in ORIGEN2

Libraries	Make New Library ?
One-grouped Cross Section Libraries	YES
Subroutines of Variable Actinide Cross Section	YES
Decay and Fission Yield	YES
Photon Spectrum (18 groups)	NO

Target reactor specifications for LWR are shown in **Table 2**. The parameters are selected from the current design of each reactor system. **Table 3** indicates the parameters for FBR libraries. These are chosen by requests from analyst of fast reactor, since no fixed specification of FBR exists especially for a commercial reactor.

3. METHOD

To make libraries for LWR, we use single pin cell models that are equivalent to the target assembly. We should evaluate applicability of our single pin cell model comparing the results with these of explicit assembly model. This comparison shows that the differences between the models are small. Especially for BWR, though assembly of BWR has complicated enrichment and void ratio distribution, the maximum difference between the two models for 70 % void ratio of STEP-II assembly, which is considered to show the largest difference, is about 10% at 50 GWd/t. This result supports the validity of this single pin cell model.

For making LWR libraries, the integrated burnup code system SWAT[4] is used. SWAT has an additional package "SWAT2ORI2" to make a subroutine for the variable actinide cross section.

To compile FBR libraries, new system is developed. This system is based on diffusion calculation of whole core using JFS3-J3.2[5]. Libraries of blanket region are prepared as well as those of core regions.

Table 2 Parameters for LWR Libraries

	PWR	BWR		
Fuel Assembly	17×17 Assembly	STEP-I Assembly	STEP-II Assembly	STEP-III Assembly
U-235 Initial Enrichment	3.4, 4.1, 4.7	2.7	3.7	4.7
Maximum Burnup(GWd/t)	55	40	40	40
Void Ratio(%)	NODATA	40	0,40, 70	40
Identifiers of Libraries	PWR34J32[1] PWR41J32 PWR47J32	BS140J32[2]	BS200J32 BS240J32 BS270J32	BS340J32

1 - Library for PWR initial U-235 enrichment is 3.4 % based on JENDL-3.2

2 - Library for BWR STEP-I assembly, void ratio is 40 % based on JENDL-3.2

Table 3 Parameters for FBR Libraries

Type of Reactors	Fuel	Initial Pu Content
Small Experimental Reactor(JOYO)	MOX	LWR
Prototype Reactor (MONJU)	MOX	LWR
600 MW Demonstration Reactor	MOX	LWR
600 MW Demonstration Reactor	MOX	FBR
600 MW Demonstration Reactor	METAL	LWR
600 MW Demonstration Reactor	NITRIDE	LWR
1300 MW Commercial Reactor	MOX	LWR
Pu Burner		

Figure 1 indicates a flowchart of compilation of the libraries. As this figure shows, an infinite diluted cross section library of 73 groups is compiled with CRECTJ5[6] using a neutron spectrum. The spectrum is prepared by connecting the standard neutron spectrum (70 groups) which was used for making JFS3-J3.2 and prompt fission neutron spectrum of Pu-239. Also, diffusion calculation is performed for target reactors with JFS3-J3.2. We obtain the neutron spectrum and effective cross section of 70 groups by that calculation. Data in the 73 groups library are collapsed to infinite diluted one grouped data with 73 group neutron spectrum. This spectrum is prepared with connecting the neutron spectrum from that diffusion calculation with JFS3-J3.2 and the prompt fission neutron spectrum of Pu-239. Finally, the infinite diluted cross

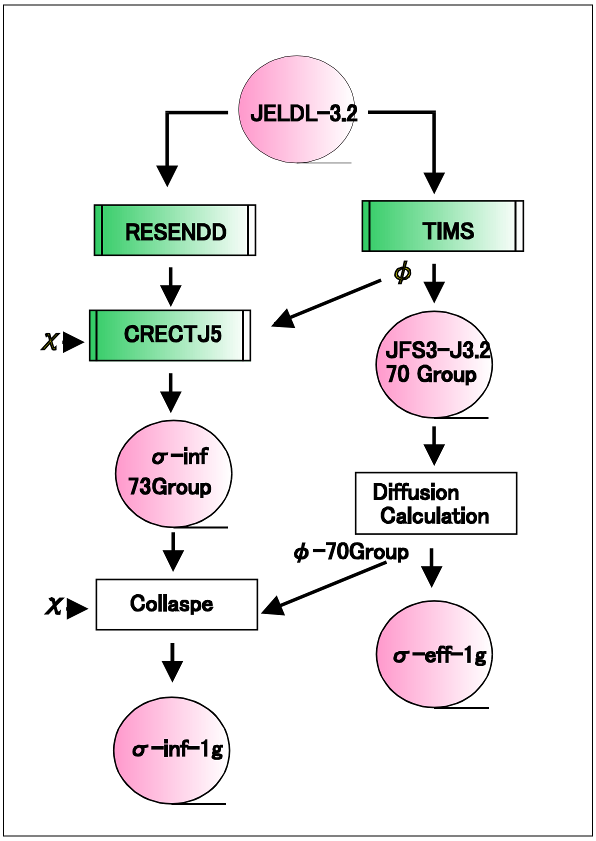


Figure 1 Flow of Compilation of Libraries for FBR

Table 4 Averaged Ratios of Calculation to Experiment(C/E) using several libraries for analysis of PIE of PWR.

	PWR-US	PWR-UE	PWR41J32
U-235	0.88	0.95	1.00
U-236	1.01	1.00	0.95
U-238	1.00	1.00	1.00
Pu-238	0.83	1.02	0.87
Pu-239	0.85	0.98	1.00
Pu-240	0.95	0.99	1.02
Pu-241	0.78	0.96	1.00
Pu-242	0.83	0.92	0.98
Am-241	0.85	1.07	1.12
Cm-244	0.66	0.92	0.82

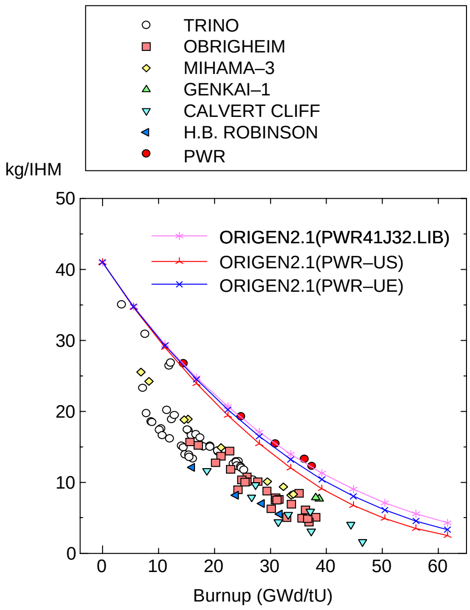
sections are substituted with the effective cross section data from that diffusion calculation. This system will be distributed for users to make a new library for satisfying each user's purpose.

The fission yield and decay libraries are also compiled based on JNDCFPV2 for each LWR and FBR systems. The decay heat values calculated by using these libraries have been recommended by the Atomic Energy Society of Japan for analysis of reactor decay heat. The direct fission yield for LWR is calculated based on the data in JNDCFPV2 considering the energy range of fission reaction evaluated by SWAT.

4. RESULTS

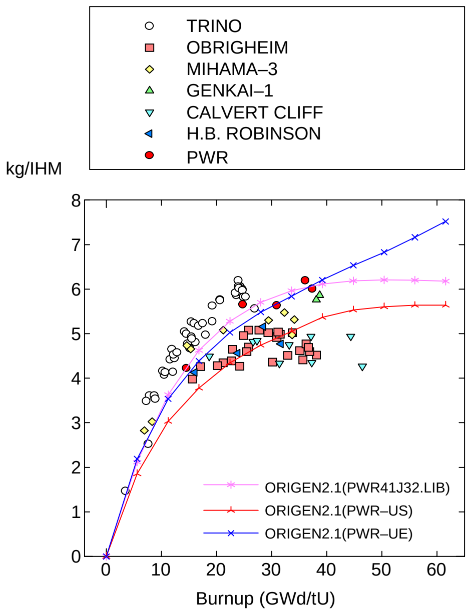
LWR Libraries

The examples of results obtained by using new library are shown in **Figure 2**, **3**, **4** and **5**, respectively for U-235, Pu-239, Pu-240 and Pu-241. In these figures, the results are compared with five PIE data for PWR 17×17 type assembly (Marks in the figures are PWR). The initial enrichment of U-235 of these samples is 4.1%. Averaged C/E values (ratio of calculated result to experimental result) of these five samples are indicated in **Table 4**. This table reveals improved results for U and Pu isotopes using our new library PWR41J32 except for Pu-238. The results of Am and Cm is also rather good. These figures and tables conclude that our libraries based on JENDL-3.2 are adequate for analysis of the current fuel.



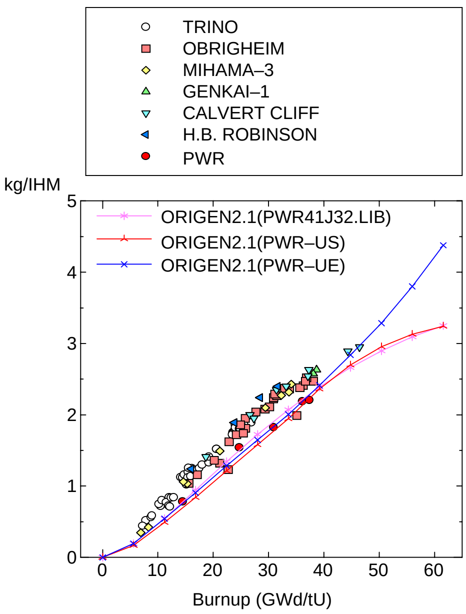
c:/ngdata/sigma/newlib/u235–971111pwrj32.ngp

Figure 2 Comparison of Results of PIE and Calculations by ORIGEN2(U-235)



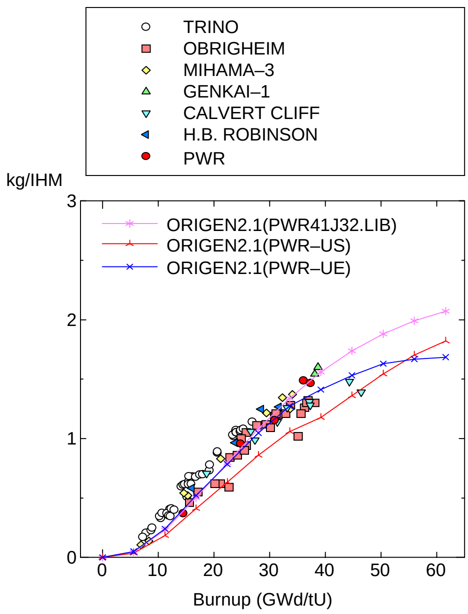
c:/ngdata/sigma/newlib/pu239–971111pwrj32.ngp

Figure 3 Comparison of Results of PIE and Calculations by ORIGEN2(Pu-239)



c:/ngdata/sigma/newlib/pu240–971111pwrj32.ngp

Figure 4 Comparison of Results of PIE and Calculations by ORIGEN2(Pu-240).



c:/ngdata/sigma/newlib/pu241–971111pwrj32.ngp

Figure 5 Comparison of Results of PIE and Calculations by ORIGEN2(Pu-241).

FBR Libraries

Since there are no disclosed PIE data, calculated amounts of generation or depletion of isotopes are compared between the old and new libraries. **Table 5** presents the comparison between them. Old Library is “LMFBR: Advanced oxide, LWR-Pu/U/U/U” in ORIGEN2. As this table shows, some isotopes have large difference of results.

Table 5 Comparison of Change of Weight(g) using Old and New Libraries for FBR.

	Old Lib.	New Lib.	N/O
U-235	1.15e+03	1.20e+03	1.05
U-238	6.99e+04	7.24e+04	1.04
Pu-238	2.09e+03	2.06e+03	0.99
Pu-239	2.60e+03	1.88e+03	0.72
Pu-240	-9.30e+02	-1.93e+03	2.08
Pu-241	9.80e+03	9.34e+03	0.95
Pu-242	4.20e+02	3.20e+02	0.76
Np-237	-3.25e+02	-2.09e+02	0.64
Am-241	-2.07e+03	-2.05e+03	0.99
Am-243	-1.23e+03	-1.45e+03	1.18
Cm-244	-1.53e+02	-3.12e+02	2.04

5. CONCLUSION

New libraries for ORIGEN2 were developed based on JENDL-3.2 using the latest core parameters. The target reactors for LWR are PWR (17×17 fuel assembly) and BWR (step I, II or III assembly). The single pin cell model is employed so as to be equivalent to the target assemblies. For PWR fuel, initial enrichment of U-235 is taken as a parameter for the libraries. For BWR fuel, initial enrichment of U-235 and axial void ratio are taken as parameters. To make libraries for LWR, the integrated burnup code system SWAT is used.

For FBR libraries, several types of cores(JOYO, MONJU, 600 MW prototype and 1300 MW commercial) and fuels (MOX, metal or nitride) are taken parameters. Libraries for not only core but also blanket region are developed. For this case, the results of core calculations using JFS3-J3.2 are compiled for making FBR libraries.

Current status of these libraries are under evaluation. After extensive tests performed, these libraries will be hopefully distributed for use in many fields of nuclear engineering.

6. Acknowledgments

Authors are deeply indebted to the members of Analytical Chemistry Laboratory of JAERI. They carried out the PIE for PWR assembly.

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