VALIDATION OF SWAT FOR BURNUP CREDIT PROBLEMS BY ANALYSIS OF PIE OF 17×17 PWR FUEL ASSEMBLY

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INTRODUCTION

For adopting burnup credit in transport or storage of spent fuel(SF), development of a reliable burnup calculation code is crucial. For this purpose, data of Post Irradiation Examination (PIE) (Naito et al., 1993) have been extensively analyzed to evaluate accuracy of burnup calculation codes for a 14 × 14 or 15 × 15 PWR fuel assembly(Suyama et al.,1994). For example, PIE at TRINO in Italy has been one of the most famous ones for this purpose. Moreover, PIE at Mihama-3 PWR in Japan was disclosed recently, and can be used for the validation of burnup codes.

However, design of current PWR fuel assembly has shifted, i.e., a 17×17 type fuel assembly is widely used. This means that we should validate our burnup codes for this new-designed assembly. From 1995, PIE for a 17×17 fuel assembly irradiated in a Japanese PWR has been made by a group of Analytical Chemistry Laboratory of Japan Atomic Energy Research Institute (JAERI). Any PIE and results of analysis have not been opened for this type of assembly in the world.

This study shows results of analysis of this latest PIE with SWAT(Suyama et al.,1996) and ORIGEN2.1(Croff,1983). SWAT is a integrated burnup code system that has been developed by Tohoku University and JAERI. The results show that SWAT can more precisely predict nuclide composition of latest PWR assembly than ORIGEN2.1.

POST IRRADIATION EXAMINATION ON 17×17 PWR FUEL ASSEMBLY

A 17×17 PWR fuel assembly was examined at Analytical Chemistry Laboratory of JAERI. The fuel assembly was irradiated in a typical 870 MWe Japanese PWR. After nondestructive γ scanning, five samples were cut off from five different positions of a fuel rod in the assembly and destructive analyses were carried out for each sample. Table 1 shows measured burnup values of each sample.

In this PIE, nuclide composition of the actinide (U, Np, Pu, Am, Cm) and fission products (Ru, Sb, Cs, Ce, Nd and Eu) was measured by the methods of α , γ and mass-spectrometry. Burnup values of these samples were determined with Nd-148 method.

Table 1 Burnup(MWd/tHM) of Each Sample

Sample No	1	2	3	4	5
Burnup (MWd/tHM)	14,678	25,238	36,700	38,064	31,363

BURNUP CODE SYSTEM SWAT

SWAT(Step-Wise Burnup Analysis Code System) is an automated driver code. Figure 1 shows the flow of analysis by SWAT.

In SWAT calculation, the total burnup history is divided into "burnup steps". SWAT calculates the neutron spectrum and effective cross sections with the SRAC code system(Okumura et al., 1996) for each burnup step, prepares a one grouped cross section library for ORIGEN2, and performs burnup calculation with ORIGEN2 using that new library.

SWAT consists of four codes: SRAC, ORIGEN2, TABMAK, and LIBMAK. SRAC is neutronics code widely used in Japanese research institutes. ORIGEN2 is one of the most famous point burnup codes. LIBMAK and TABMAK are functional codes: LIBMAK prepares libraries for ORIGEN2 using a calculated result of SRAC. TABMAK prepares input data for SRAC and ORIGEN2 from the result of ORIGEN2 for next burnup step. Moreover, SWAT contains cross section library based on JENDL-3.2(Nakagawa et al., 1995), and decay and fission yield library

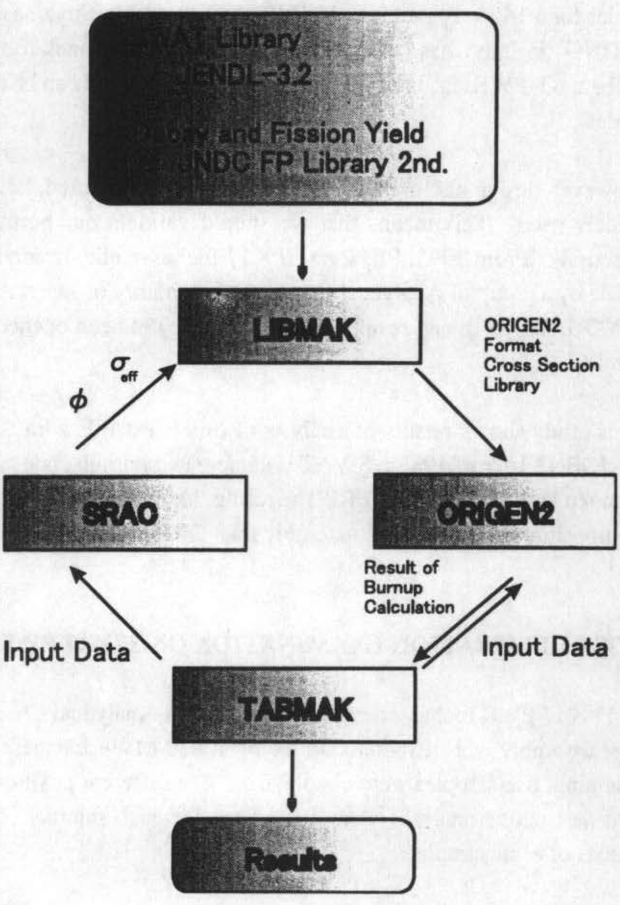

Figure 1 Flow of Calculation in SWAT

based on JNDC FP library second version(Tasaka et al., 1990).

This means that we can carry out burnup calculation of more than thousands of isotopes by ORIGEN2 using the latest cross section library prepared by SRAC considering changes in the neutron spectrum and effective cross section during the burnup.

DETAILS OF ANALYSIS

SWAT was used for the analysis of this PIE. For SWAT, a one-dimensional pin cell model was adopted, which has geometrical features averaged over the entire assembly. Temperature and density of coolant were determined by the axial position of each sample. Though the detailed history of irradiation of each sample was not opened, it was known that this fuel rod was irradiated under averaged condition. For this reason, a constant power level is approximated in this analysis. For used data libraries, cross section library based on JENDL-3.2, and fission yield and decay constants based on JNDC FP Library second version were used.

For comparison, ORIGEN2.1, the latest version on ORIGEN2 was also used for the analysis. The PWR-US library was used by ORIGEN2.1. The same power histories with the case of SWAT are used.

RESULTS OF ANALYSIS

Tables 2 and 3 show values of C/E (a ratio of a calculated value by SWAT and ORIGEN2.1 to the experimental one). These tables also show averaged C/E values for each isotope.

For actinide-only burnup credit, the accuracy of calculation for U and Pu is most important. From this point of view, SWAT shows good results. The relative deviations of calculation from experiment are less than 5% for U and Pu isotopes except Pu-238. Namely, the averaged C/E values of isotopic concentrations are 0.99, 0.99, 1.02, and 0.96 for U-235, Pu-239, Pu-240 and Pu-241, respectively.

Analysis by means of ORIGEN2.1 with PWR-US library indicates larger deviations than those of SWAT. Deviations between experiment and calculation are more than 10% for U and Pu isotopes, Pu-239 and Pu-240 particularly show more than 30%: The averaged values of C/E are 0.90, 0.77, 0.76, and 0.92 for U-235, Pu-239, Pu-240 and Pu-241.

For higher actinide, values of C/E by SWAT and ORIGEN2.1 are worse than the case of U and Pu, SWAT however shows improved results than ORIGEN2.1 with PWR-US library.

Table 2 Result (C/E) by SWAT

SAMPLE	1	2	3	4	5	Averaged
U-234	1.02	0.93	1.19	1.18	0.87	1.04
U-235	1.00	1.00	0.98	0.98	0.98	0.99
U-236	0.96	0.94	0.96	0.96	0.95	0.95
U-238	1.00	1.00	1.00	1.00	1.00	1.00
Pu-238	0.81	0.76	0.87	0.87	0.82	0.83
Pu-239	1.03	0.98	0.98	0.97	0.98	0.99
Pu-240	1.03	1.00	1.03	1.03	1.03	1.02
Pu-241	1.00	0.94	0.96	0.96	0.96	0.96
Pu-242	0.96	0.92	0.96	0.97	0.96	0.95
Am-241	0.76	1.04	1.05	1.43	0.99	1.05
Am-242m	0.68	0.68	0.69	0.70	0.71	0.69
Am-243	0.86	0.86	0.93	0.94	0.89	0.90
Cm-242	0.76	0.67	0.62	0.54	0.79	0.68
Cm-243	0.60	0.56	0.67	0.65	0.57	0.61
Cm-244	0.77	0.67	0.81	0.79	0.78	0.76
Cm-246	0.39	0.45	0.76	0.74	1.10	0.69
Ru-106	1.00	1.19	1.30	1.31	1.12	1.18
Sb-125	1.80	1.74	2.15	2.62	2.00	2.06
Cs-134	0.87	0.84	0.88	0.88	0.86	0.87
Cs-137	1.00	0.99	1.00	1.00	1.00	1.00
Ce-144	1.01	1.02	1.00	1.09	1.03	1.03
Nd-142	0.78	0.91	0.85	0.87	0.94	0.87
Nd-143	0.98	0.98	0.97	0.98	0.98	0.98
Nd-144	0.99	0.99	1.01	0.97	1.00	0.99
Nd-145	1.01	1.01	1.02	1.02	1.02	1.02
Nd-146	1.02	1.01	1.01	1.01	1.01	1.01
Nd-148	1.01	1.01	1.02	1.02	1.02	1.02
Nd-150	0.98	0.99	0.99	0.99	1.00	0.99
Eu-154	0.94	0.88	0.92	0.89	0.90	0.91

Table 3 Results(C/E) by ORIGEN2.1

Sample	1	2	3	4	5	Average
U-234	1.05	0.96	1.23	1.20	0.88	1.06
U-235	0.97	0.93	0.84	0.85	0.91	0.90
U-236	1.01	1.00	1.01	1.01	1.00	1.01
U-238	1.00	1.00	1.00	1.00	1.00	1.00
Pu-238	0.69	0.68	0.86	0.92	0.86	0.80
Pu-239	0.84	0.74	0.73	0.77	0.78	0.77
Pu-240	0.90	0.75	0.69	0.73	0.75	0.76
Pu-241	0.80	0.96	0.92	0.91	1.03	0.92
Pu-242	0.79	0.91	1.06	1.07	1.08	0.98
Am-241	0.61	0.93	0.95	1.34	1.01	0.97
Am-242m	1.20	1.41	1.62	1.79	1.90	1.58
Am-243	0.66	0.77	1.04	1.13	1.02	0.92
Cm-242	0.52	0.54	0.57	0.51	0.76	0.58
Cm-243	0.50	0.54	0.78	0.84	0.71	0.67
Cm-244	0.50	0.49	0.79	0.88	0.80	0.69
Cm-246	0.21	0.27	0.58	0.66	0.91	0.53
Ru-106	0.90	1.10	1.25	1.28	1.09	1.12
Sb-125	3.14	2.98	3.67	4.52	3.47	3.56
Cs-134	0.84	0.84	0.92	0.95	0.92	0.89
Cs-137	0.97	0.96	0.97	0.97	0.98	0.97
Ce-144	1.02	1.02	1.00	1.09	1.02	1.03
Nd-142	1.01	1.22	1.19	1.23	1.29	1.19
Nd-143	0.98	0.97	0.94	0.95	0.97	0.96
Nd-144	1.00	1.02	1.06	1.01	1.02	1.02
Nd-145	1.02	1.02	1.02	1.02	1.02	1.02
Nd-146	1.02	1.01	1.02	1.02	1.02	1.02
Nd-148	1.02	1.01	1.01	1.01	1.02	1.01
Nd-150	0.96	0.98	0.98	0.99	1.00	0.98
Eu-154	1.05	1.09	1.32	1.36	1.30	1.22

Finally, for FP shown in Table 2 and Table 3, values of C/E are almost the same between SWAT and ORIGEN2.1. However results of Sb-125 and Eu-154 are improved by SWAT than ORIGEN2.1.

The reasons stated above conclude that reliability of SWAT with JENDL-3.2 library to predict isotopic composition of the irradiated 17×17 PWR fuel assembly is higher than ORIGEN2.1 with PWR-US library.

CONCLUSION

In this study, PIE of the latest type PWR fuel assembly was shown. This PIE is very important since there is no other similar disclosed PIE in the world. The results of this PIE were analyzed with SWAT. SWAT is the driver code for burnup calculation using the latest nuclear data, considering change of the neutron spectrum and effective cross sections during the burnup. ORIGEN2.1 is also used for comparison. The results of analysis show that SWAT has improved ability to predict the isotopic concentrations of spent fuel than ORIGEN2.1 does for the 17 × 17 PWR fuel assembly.

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REFERENCES

Croff, ORIGEN2:A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials, Nucl. Technol., 62,pp335-352(1983).

Naito et al., Databook of the Isotopic Composition of Spent Fuel in Light Water Reactors,JAERI-M 93-061(1993).

Nakagawa et al., Japanese Evaluated Nuclear Data Library Version 3 Revision-2: JENDL-3.2, J. Nucl. Sci. Technol., 32,pp1259-1271(1995).

Okumura et al., SRAC95; General Purpose Neutronics Code System, JAERI-Data/Code 96-015(1996).

Suyama et al., Analysis of Post Irradiation Experiments in PWR Using New Nuclear Data Libraries, J. Nucl. Sci. Technol., 31, pp596-608(1994).

Suyama et al., Improvement of Burnup Code System SWAT for Use in Burnup Credit Problem, In proceedings of PHYSOR96, Vol. 4, ppL53-62, Mito, Japan, September 16-20, 1996.

Tasaka et al., JNDC Nuclear Data Library of Fission Products - second version -, JAERI-1320(1990).