Requests on Nuclear Data in the Backend Field through PIE Analysis

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1. Introduction

The working group on evaluation of nuclide generation and depletion is active in response to requests from the industry group in the backend field.

Accurate evaluations of fuel composition, neutron/gamma production, radioactivity and decay heat for spent fuel are requested in the backend field. The ORIGEN2 code is widely used to evaluate these quantities in the industry. This WG, therefore, prepares ORIGEN2 libraries for LWR and FBR based on JENDL-3.2 reflecting user requests[1]. Fuel compositions for spent fuels are basic quantities to be used in the criticality safety analysis and radiation shielding. So, the accuracy of fuel compositions for LWR and FBR spent fuels have been investigated through PIE analysis using recent nuclear data such as JENDL-3.2.

Hereafter, the present status of fuel composition analysis for LWR and FBR (fast reactor spent fuel) and the requests for nuclear data are described. After that, the reason of discrepancies between measured and calculated values will be studied by the sensitivity analysis.

2. Present status of fuel composition analysis for LWR and fast reactor spent fuels

2.1 Fuel composition analysis for LWR spent fuels

In JAERI, spent fuel composition irradiated in commercial PWR and BWR have been measured and analyzed by burn-up code[2].

Irradiation data on measured fuels are shown in Table 1 and measured nuclides are shown in Table 2.

Results of fuel composition analyses by burn-up code SWAT (SRAC-ORIGEN2) are shown in Fig. 1, in which nuclear data are JENDL-3.2+JNDC-V2 and unit-cell models mock-up the sample rods considering actual irradiation histories are used.

Accuracy of analyses for U and Pu isotopes are quite good ($|C/E-1| < 5\%$). But, errors for Am-241 are +10～20% and those for major neutron production nuclide (Cm-242 and Cm-244) are large (-15～50% for Cm-242, -23~26% for Cm-244).

On the other hand, accuracy of analyses for fission products are good ($|C/E-1| <$ 10%) except Sm-152.

From the above results, we have the following remarks..

・It is possible to reduce large margin assumed in the present criticality analysis for spent fuels based on the burnup code considering actual irradiation histories

・It is desired that the insufficient accuracy for Cm242 and Cm244 will be improved

2.2 Fuel composition analysis for fast reactor spent fuel

Post irradiation examination (PIE) analysis of for the fast reactor JOYO mixed oxide spent fuel has been carried out. The outline of the measured spent fuel is shown in Table 3. It has been originally irradiated at the 2nd row and it was later moved to the

4th row and irradiation has been continued until the fuel burn-up reached approximately 58.2GWd/Mt. The irradiation position of the measured spent fuel and PIE positions are shown in Fig. 2. The PIE has been carried out for three fuel pins (No.7, 76, C1). Axial positions examined are the core center height, as well as the upper and lower ends of the fuel region.

Burn-up composition is calculated using ORIGEN2 code. One-group cross-section was collapsed from the 73 group constant set based on JENDL-3.2, using the neutron spectrum of each PIE position, which was calculated using CITATION. The neutron flux used as an input to the ORIGEN2 was calculated by the JOYO core management code system.

The comparison between the calculated and measured burn-up composition is shown in Fig. 3. Calculated results of U, Pu and ^{148}Nd agree well with measured values. Am isotopes, ^{242}Cm and ^{244}Cm are overestimated, and ^{237}Np and ^{243}Cm are underestimated. The reason for the disagreement is understood to be that the capture cross-section of Am isotopes are underestimated, and that of ^{237}Np and ^{243}Cm are overestimated.

3. Studied on capture cross sections for TRUs

Large errors for TRUs capture cross section on JEF2.2 were indicated from the CEA experiment mock-up thermal and epi-thermal reactor[3]

Discrepancies for capture cross sections of heavy nuclides between APOLLO-1 calculations using JEF-2.2 and experiments are shown in Table 4.

The comparison for capture cross section of TRUs between JENDL-3.2 and JEF-2.2 is shown in Table 5.

Errors for capture cross sections of TRUs in JENDL-3.2 assumed from the combination of Table 4 and 5 are shown in Table 6.

We study through PIE analysis with cross section considering assumed error for Am-241 capture cross section shown in Table 5. The analysis by burn-up code MCNP-ORIGEN2[4] has been performed for the one of BWR sample rods shown in Table 1, in which JENDL-3.2+JNDC-V2 is used and analytical model is BWR assembly model with reflective boundary condition shown in Fig.4.

To investigate the effect of error on Am-241 capture cross section, we perform the calculation with +25% correction for Am-241 capture cross section at all burnup steps in addition to the no correction calculation.

The results of comparison for actinide compositions between two calculations are shown in Fig.5 for U-isotopes, Fig. 6 for Np and Pu-isotopes and Fig.7 for Am and Cm-isotopes.

The effect to actinide compositions caused by the correction of Am-241 capture cross section are

- U-234～238：very small (<0.3%)
- Np-237：small(～3%)
- Pu- 238～242 ：small (<1%)
- Am-241：large (-15%)
- Am-242,242m：large (+7%)
- Cm-242：large (+9%)
- Cm-244：small (+0.1%)

Accuracy of TRUs compositions in the analysis for LWR spent fuels considering the above effects are summarized as follows.

The overestimation of Am-241(10～20%) is well solved, but the underestimations of Cm--242 (-15～50%) and Cm244 (-23～26%) are solved partly in the analysis using the +25% correction for Am241 capture cross section in JENDL-3.2.

We, therefore, wish that the improvement of cross sections for actinides beyond Pu are very important to improve the accuracy of PIE analysis.

4. Conclusion

From studies on nuclear data related to the backend field based on activities of our WG, we have the following remarks..

- The burnup codes used in our WG such as SRAC, SWAT and ORIGEN with cross sections considering actual neutron spectrum has been verified through PIE analysis based on JENDL-3.2
- It is possible to reduce large margin assumed in the present criticality analysis for spent fuels using the verified burnup codes
- It is desired that the nuclear data files for Cm-242 and Cm-244 will be revised to improve the accuracy of PIE analysis

Our wishes is to use improved nuclear data to be solved the problems in our studies.

Acknowledgment

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References

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Table.1 Irradiation Data on Measured LWR Fuels

Reactor Type	BWR	PWR
Irradiation Plant	Fukushima-2	Takahama-3
Burnup (GWd/Mt)	4～44	8～48
Void History (%)	0～73	—
No, of Data	17	16

Table.2 Measured Nuclides for LWR Fuels

	Actinide (18 Nuclides)		Fission Product (17 Nuclides)
Element	Nuclides	Element	Nuclides
U	U234,U235,U236,U238	Cs	Cs134, Cs137
Np	Np237	Ce	Ce144
Pu	Pu238,Pu239,Pu240,Pu241, Pu242	Nd	Nd143,Nd144,Nd145,Nd146,Nd148, Nd150
Am	Am241,Am242m,Am243	Eu	Eu154
Cm	Cm242,Cm243,Cm244,Cm245, Cm246	Sm	Sm147,Sm148,Sm149,Sm150,Sm151 ,Sm152,Sm154

Table 3 Outline of the Measured Spent Fuel

MOX Fuel Content	U-235 Enrichment:18.6wt%	
	Pu Content:28.5wt%	
Irradiation Condition	Core Resident Period:Dec.14,1990 ～Sep.24,1997	
	Address:2nd Row(2B1) Irradiaion ϕ_{total}: 3.3×10^{15} n/cm^2 s (Subassembly	Address:4th Row(4D1) Irradiation ϕ_{total}: 2.1×10^{15}n/cm^2 s (Subassembly
F_{total} (Subassembly)	9.9×10^{22} n/cm^2	
Burn-up (Subassembly Averaged)	58.2GWd/Mt	

Table 4 %-(C/E-1) for capture cross sections of heavy nuclides between APOLLO-1 calculations using JEF-2.2 and experiments

Nuclide	SHERWOOD	ICARE/S
U238	+1.1±1.8	+5.3±2.5
Pu238	—	+16.0±12.0
Pu239	+2.6±1.6	+1.0±8.0
Pu240	+1.9±1.6	+0.1±3.2
Pu241	+5.6±10.0	-4.7±6.1
Pu242	-0.3±3.2	-12.3±6.4
Am241	-20.0±15.0	-20.0±11.0
Am243	-22.4±5.0	—
Cm244	+7.9±12.3	—

Value：(C/E-1)±2σ in %
SHERWOOD：Square lattice experiment mock-up thermal reactor
ICARE/S ：Tight lattice experiment mock-up epi-thermal reactor

Table 5 Comparison for TRUs capture cross section between JENDL-3.2 and JEF2.2

	Thermal (2200m/sec)			Resonace Integral		
Nuclide	JENDL-3.2	JEF-2.2	JENDL/JEF	JENDL-3.2	JEF-2.2	JENDL/JEF
Am241	600.4	616	0.97	1305	1450	0.90
Am243	78.50	75.94	1.03	1823	1810	1.01
Cm244	15.10	14.41	1.05	660	634	1.04

Where, cross sections are in barns

Table 6 %-errors for capture cross sections of TRUs in JENDL-3.2 assumed from the combination of Table 3 and 4

Nuclide	Thermal	Epi-ithermal
Am241	-23.0±15.0	-30.0±11.0
Am243	-21.7±5.0	—
Cm244	+8.3±12.3	—

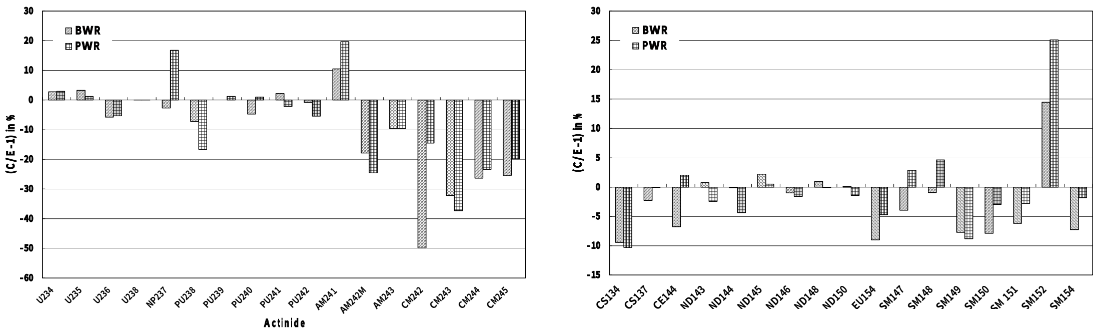


Fig.1 Results of SWAT Analyses for LWR Spent Fuels based on JENDL-.3.2

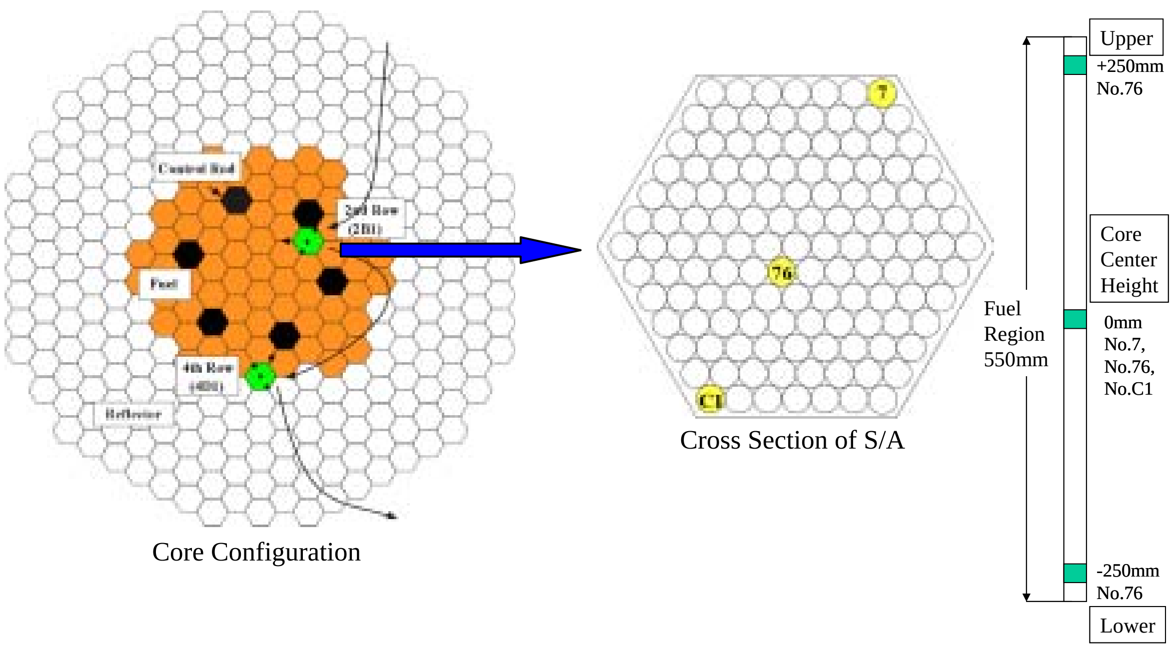


Fig.2 Irradiation Position and PIE Position of JOYO MK-II Measured Fuel

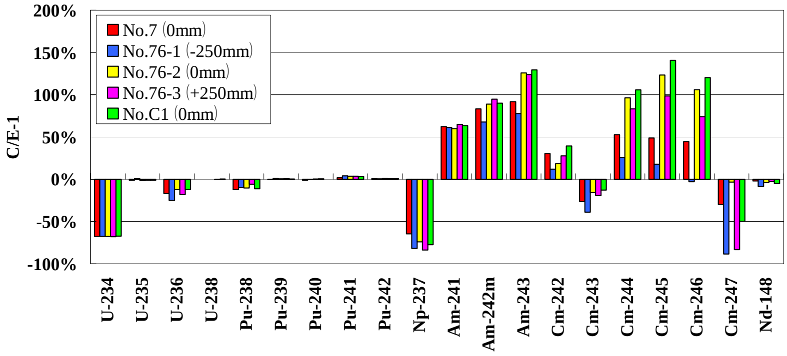
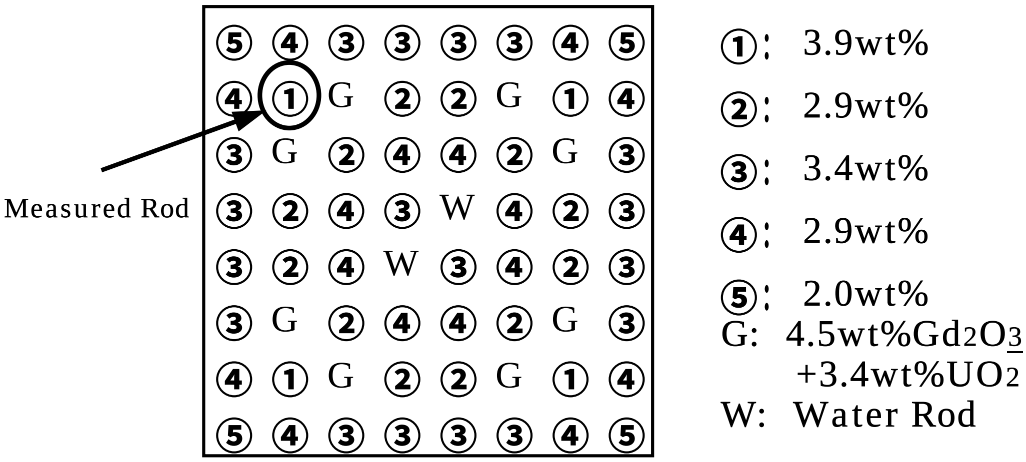


Fig.3 Comparison of Calculated and PIE Results of JOYO MK-II Spent Fuel



BWR 8x8 Assembly

Historical Void = 3% (Lower Cross Section)

Discharged Assembly Burnup = 32.5 GWd/Mt

Fig.4 Analytical Model in MCNP-ORIGEN Burnup Calculation for BWR sample rod

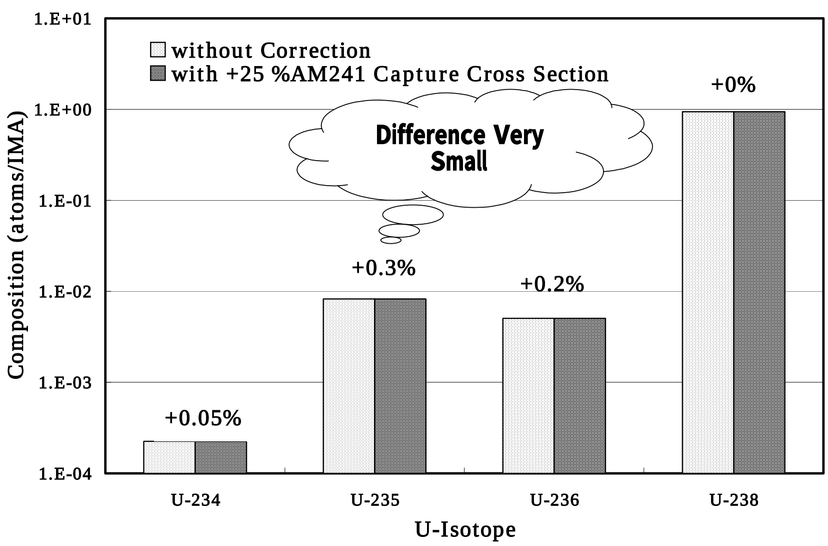


Fig.5 The comparison of vo,positions for U-isotopes between two calculations

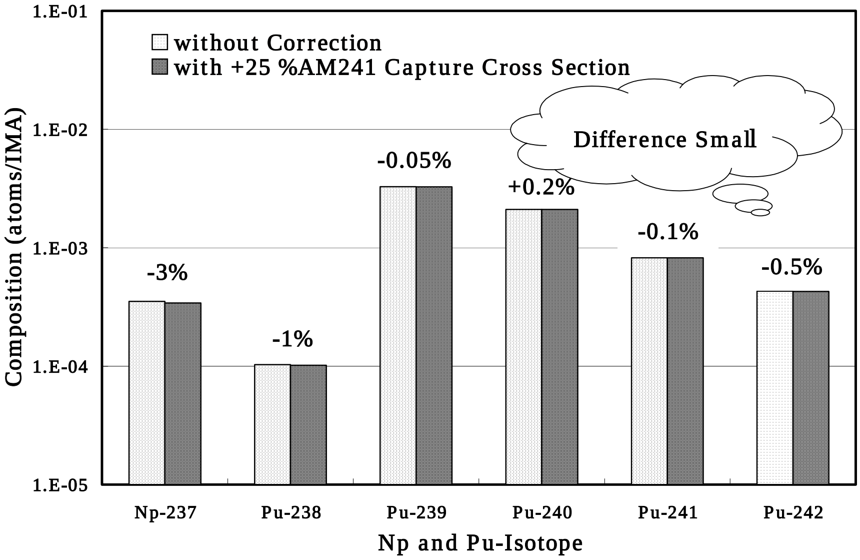


Fig.6 The comparison of compositions for Np&Pu-isotopes between two calculations

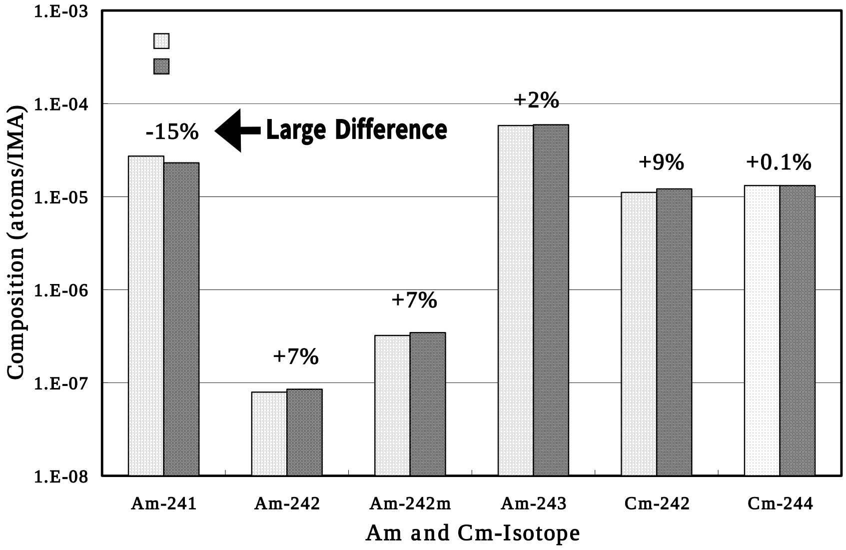


Fig.7 The comparison of compositions for Am&Cm-isotopes between two calculations