ANL-ARC-245

Nuclear Data: Progress Report on Sensitivity Analysis at ANL in FY2012. The TRAPU Experiment

Nuclear Engineering Division

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ABSTRACT

In the framework of the Nuclear Data Uncertainty Reduction project, one of the major activities assigned to ANL for FY2012 is related to the sensitivity study to be conducted for 82 isotope transmutations in specific samples of the PROFIL-1, PROFIL-2 and TRAPU irradiation experiments. The analysis of these irradiation experiments for the cross section adjustment of the Nuclear Data Uncertainty Reduction project is primarily intended to assess the experimental database on minor actinides made available by the CEA (France) and to validate and improve the current set of minor actinide cross sections and decay constants. The required sensitivity analyses for the PROFIL-1 and PROFIL-2 experiments have been described in a previous report **[1]** distributed in June 2012. This document presents the now completed analyses on the TRAPU experiment.

Calculations were performed with the ANL depletion perturbation theory code DPT, which is able to evaluate the impact of the flux distribution change during irradiation on the nuclide density evolution. This impact, though commonly neglected in previous studies based for instance on the ERANOS and NUTS codes, was demonstrated to be non-negligible.

The sensitivities obtained for 44 isotope buildups in the TRAPU irradiated pins have been delivered to the other participants in the Nuclear Data Uncertainty Reduction project. A global database is now being created at ANL that is expected to contain in a specific format all sensitivities and C/Es of the several experiments investigated by the different Labs (primarily, ANL and INL) involved in the Nuclear Data Uncertainty Reduction project, as well as detailed information on the codes, models and approximations used for the calculations.

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1 Introduction

The primary objectives of the Nuclear Data Uncertainty Reduction project **[1,2,3,4,5]** are to minimize the impact of nuclear data uncertainties in reactor core calculations and to provide accurate nuclear data needs assessments by incorporating science based covariance data and key fast spectrum integral data in the sensitivity analyses, providing the best broad covariant constraints (inter-reaction and inter-isotope) currently available. The collaborative efforts of the laboratories involved in this work project have made nice advances toward these objectives.

Within the project, the ANL activities include: (1) generation of nuclear data sensitivity coefficients, (2) study of the methods and modeling approximations for integral parameter sensitivities, (3) uncertainty propagation with new covariance data to provide feedback to the nuclear data evaluators, (4) generation of calculational models, calculated values and their uncertainties, and experimental values and their uncertainties, (5) development of data adjustment methodologies, and (6) preliminary application of these cross section adjustment methods.

The advanced nuclear systems and associated fuel cycles need good quality cross section data to provide a reliable assessment of their performance. Basic evaluated data are available for transuranic (TRU) isotopes (up to Cf) but validation is needed in order to quantify their reliability. This is traditionally done through the use of differential and integral experiments, and uncertainty assessment. Integral experiments in reactors play an essential role in nuclear data validation and improvements. The information on minor actinides (MA) that can be gathered from experiments mainly comes from small sample irradiation, reactivity oscillation, and fission and capture rate measurements. Separate isotope sample and fuel pin irradiations in power reactors provide a unique source of very useful measurements.

In the framework of the Nuclear Data Uncertainty Reduction Project started in FY2008, ANL is collaborating with other laboratories (BNL, LANL and INL) with the aim of providing feedback to nuclear data evaluators for improving the quality of basic files, and assessing their impact on advanced fuel cycles. Several experiments are being analyzed for this purpose. In order to validate and improve the current set of cross sections and decay constants of minor actinides, efforts have also been made to exploit the experimental database on MA made available by the CEA (the French Atomic Energy Commission). It was assessed that the cleanest and most useful irradiation experiments available in the CEA database are the PROFIL and TRAPU programs **[6, 7, 8]**.

At the end of 2003, ANL received from CEA detailed information about the PROFIL-1, -2, and TRAPU experiments. This includes detailed descriptions of subassemblies and experimental device locations, the power and configuration histories for the different cycles of irradiation, and three-dimensional core models. This is the result of very time consuming efforts by French colleagues that include the retrieval of information from very old files (e.g., the PROFIL-1 irradiation was performed in 1974).

A preliminary data adjustment was previously performed with the use of PROFIL and TRAPU experiments **[9, 10, 11]**, providing useful indications of trends for improved cross section evaluation. Within the current Nuclear Data Uncertainty Reduction project, it is intended to improve the previous study with the use of the most recent nuclear data evaluation (based on the ENDF/B-VII library) and of more detailed calculation models, including Monte Carlo simulations, which allow the explicit representation of pins, experimental devices, etc. Additionally, a more sophisticated sensitivity analysis on the nuclide density variation during irradiation cycles is to be performed using the depletion perturbation theory code system (REBUS-3/DPT) of ANL, which is able to evaluate the impact of the flux distribution change during irradiation on the nuclide density evolution. These effects, though commonly neglected in previous studies based for instance on the ERANOS and NUTS codes, were demonstrated to be non-negligible.

An analysis of the PROFIL and TRAPU experiments was performed by INL using MCNP5 in order to produce the best possible C/E values. For FY2012, ANL was in charge of the sensitivity study to be conducted for 82 isotope transmutations in specific samples of the PROFIL-1, PROFIL-2 and TRAPU irradiation experiments. The required sensitivity analyses for the PROFIL-1 and PROFIL-2 experiments have been described in a previous report **[1]** distributed in June 2012. This document presents the completed analyses on the TRAPU experiment.

As for the case of PROFIL-1 and PROFIL-2, the sensitivities obtained for 44 isotope buildups in the TRAPU irradiated pins have been delivered to the different laboratories involved in the Nuclear Data Uncertainty Reduction project. A global database is now being created at ANL that is expected to contain in a specific format all sensitivities and C/Es of the several experiments investigated by the different Labs (primarily, ANL and INL) involved in the Uncertainty Reduction project, as well as detailed information on the codes, models and approximations used for the calculations.

The computational methods employed and the TRAPU irradiation experiment are briefly described in **Sections 2** and **3**, respectively The sensitivity study is discussed in **Section 4** and conclusions are summarized in **Section 5**.

2 Computational Methods and Models

For the required sensitivity study of the irradiation experiments, cross sections were generated in the 33-energy group structure shown in **Table I** using the MC^2-3 code **[12]** and the ENDF/B-VII.0 nuclear data **[13]**. Depletion calculations were performed using the fast reactor fuel cycle analysis code REBUS-3 **[14]**. The flux calculations used the finite difference diffusion theory option of the DIF3D code **[15]**.

Table I. 33-Energy Group Structure of Multigroup Cross Sections

Group	Upper bound, MeV	Group	Upper bound, MeV	Group	Upper bound, MeV
1	1.41910E+01	12	6.73795E-02	23	3.04325E-04
2	1.00000E+01	13	4.08677E-02	24	1.48625E-04
3	6.06531E+00	14	2.47875E-02	25	9.16609E-05
4	3.67879E+00	15	1.50344E-02	26	6.79041E-05
5	2.23130E+00	16	9.11882E-03	27	4.01690E-05
6	1.35335E+00	17	5.53084E-03	28	2.26033E-05
7	8.20850E-01	18	3.35463E-03	29	1.37096E-05
8	4.97871E-01	19	2.03468E-03	30	8.31529E-06
9	3.01974E-01	20	1.23410E-03	31	4.00000E-06
10	1.83156E-01	21	7.48518E-04	32	5.40000E-07
11	1.11090E-01	22	4.53999E-04	33	4.14000E-07

The depletion calculation for each region was performed with the average of the beginning and end of time interval fluxes. The end of time interval flux was iteratively computed by iteration on the final nuclide densities. Burnup chains were modeled as shown in **Table II**. Capture, (n,2n), and fission reactions were considered for all actinide isotopes from U234 to Cm246. In the capture and (n,2n) reactions, short-lived intermediate products were neglected. For example, the products of capture reactions of U238, Pu242, and Am243 were represented by Pu239, Am243, and Cm244, respectively. The capture reaction of Am241 was modeled to yield Cm242, Am242m, and Pu242 with yield fractions of 0.6616, 0.20, and 0.1384, respectively. The products of (n,2n) reactions of Pu238 and Am241 were respectively represented by Np237 and Pu240. The (n,2n) reaction of Am243 was assumed to yield Am242m, Pu242, and Cm242. Cm242 was assumed to yield Am241 in 99% of its (n,2n) reactions and Np237 in 1%. The end products which are not included in the chains were represented by a fictitious dummy isotope (DUMP in **Table II**).

Important α and β decays of actinide isotopes were also considered. Specifically, α decay was considered for all actinide isotopes except for Np238 and Pu241. The β^- decays of Pu241 and Am242m and the β^+ decay of Am242m were also included in the burn chains. The decay constants in **Table II** are based on the ENDF/B-VII.0 library and provided by BNL. Fission products were modeled with five lumped fission products. The 33-group cross sections of these lumped elements were generated by weighting the 33-group cross sections of 180 fission products with fission yields of U235, U238, Pu239, Pu240, and Pu241, respectively. For full reactor depletion calculations, the lumped fission products of U234, U235, U236 and Pu236 were represented by those of U235, while the fission products of U238, Np237, Np238, and Pu238 were represented by those of U238. The fission products of Pu241 and higher actinides were represented by those of Pu241.

Table II. Burn Chains used for REBUS-3 Depletion Calculations (Excluding Fission Reactions)

Isotope	Reaction	Product	Branch ratio	Reaction	Product	Branch ratio	Decay	Product	Decay constant (1/sec)
U234	(n,γ)	U235		(n,2n)	DUMP		α	DUMP	8.946848E-14
U235	(n,γ)	U236		(n,2n)	U234		α	DUMP	3.119966E-17
U236	(n,γ)	Np237		(n,2n)	U235		α	DUMP	9.378539E-16
U238	(n,γ)	Pu239		(n,2n)	Np237		α	U234	4.915972E-18
Np237	(n,γ)	Pu238		(n,2n)	Pu236	0.346	α	DUMP	1.024465E-14
					U236	0.374			
					DUMP	0.280			
Pu236	(n,γ)	Np237		(n,2n)	DUMP		α	DUMP	7.685273E-09
Pu238	(n,γ)	Pu239		(n,2n)	Np237		α	U234	2.504505E-10
Pu239	(n,γ)	Pu240		(n,2n)	Pu238		α	U235	9.110133E-13
Pu240	(n,γ)	Pu241		(n,2n)	Pu239		α	U236	3.347745E-12
Pu241	(n,γ)	Pu242		(n,2n)	Pu240		α	Am241	1.537055E-09
Pu242	(n,γ)	Am243		(n,2n)	Pu241		α	U238	5.857252E-14
Am241	(n,γ)	Am242m	0.2000	(n,2n)	Pu240		α	Np237	5.077332E-11
		Cm242	0.6616						
		Pu242	0.1384						
Am242m	(n,γ)	Am243		(n,2n)	Am241		β^-	Cm242	1.282476E-10
							β^+	Pu242	2.682809E-11
							α	Pu238	7.009951E-13
Am243	(n,γ)	Cm244		(n,2n)	Am242m	0.5000	α	Pu239	2.980266E-12
					Pu242	0.0865			
					Cm242	0.4135			
Cm242	(n,γ)	Cm243		(n,2n)	Am241	0.99	α	Pu238	4.927852E-08
					Np237	0.01			
Cm243	(n,γ)	Cm244		(n,2n)	Cm242		β^+	Am243	2.188903E-12
							α	Pu239	7.526053E-10
Cm244	(n,γ)	Cm245		(n,2n)	Cm243		α	Pu240	1.213509E-09
Cm245	(n,γ)	Cm246		(n,2n)	Cm244		α	Pu241	2.584066E-12
Cm246	(n,γ)	DUMP		(n,2n)	Cm245		α	Pu242	4.614398E-12

The sensitivity coefficient calculations were performed using the depletion perturbation theory code DPT **[16]**. The depletion perturbation method implemented in DPT is based on the non-equilibrium and equilibrium fuel cycle analysis methodologies of REBUS-3. The code calculates the sensitivity coefficients of a given response functional with respect to cross sections and initial nuclide densities. The adjoint nuclide density equations are solved using iteration methods similar to those used for solving the nuclide density equations, and the generalized adjoint flux equations are solved using DIF3D. DPT currently generates sensitivity coefficients for five responses: the beginning and end of cycle k-effective values, the burnup reactivity swing, the power fraction in any core region, and the end of cycle mass of any isotope in any core region. Currently, the flux solution option is limited to the finite difference diffusion theory option in RZ geometry, and the burn cycle is modeled with only a single time interval. A brief overview of the depletion perturbation equations solved by DPT was given in the previous report **[1]**. It is recalled that DPT has computational capabilities to decompose the sensitivities of nuclide density evolution during burnup in their separate effects of direct, number density, flux, adjoint flux, and power terms. Specifically, the nuclide density term represents the response parameter increase through the changes in nuclide transmutation rates caused by the increase in a cross section of interest, the flux terms represent the response increase through the changes in the neutron flux distribution in space and energy for a fixed flux level, and the power terms represent the response increase through the changes in the neutron flux level for a given power level.

3 The TRAPU Irradiation Experiment

The TRAPU experiment consisted of a six-cycle irradiation (10^{th} to 15^{th}) of mixed-oxide pins that contained plutonium of different isotopic compositions but heavily charged in the higher isotopes (Pu240, Pu241 and Pu242) compared to typical Phenix fuel. Standard pins were placed in regular Phenix assemblies (CPa 319 and CPa 321) located in the third row of the reactor shown in **Figure 1**. Three types of pins with different plutonium vectors were used as shown in **Figure 2**: TRAPU-I, TRAPU-II (2 pins placed in the positions 93 and 125), and TRAPU-III (2 pins placed in the positions 92 and 126).

The cycle lengths and the corresponding reactor power that were provided in the CEA specifications of the TRAPU irradiation experiment are provided in **Table III**.

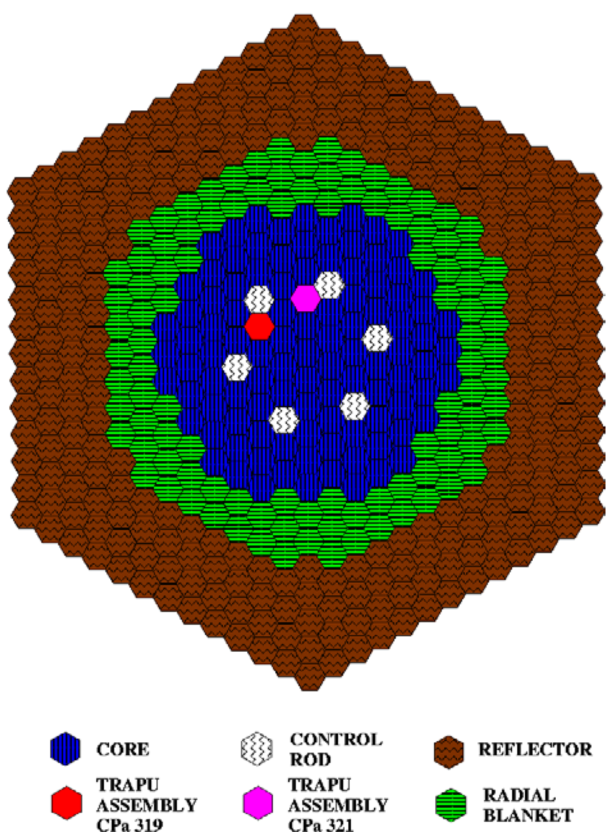

Figure 1. Positions of the TRAPU Assemblies in the Phenix Fast Reactor Core

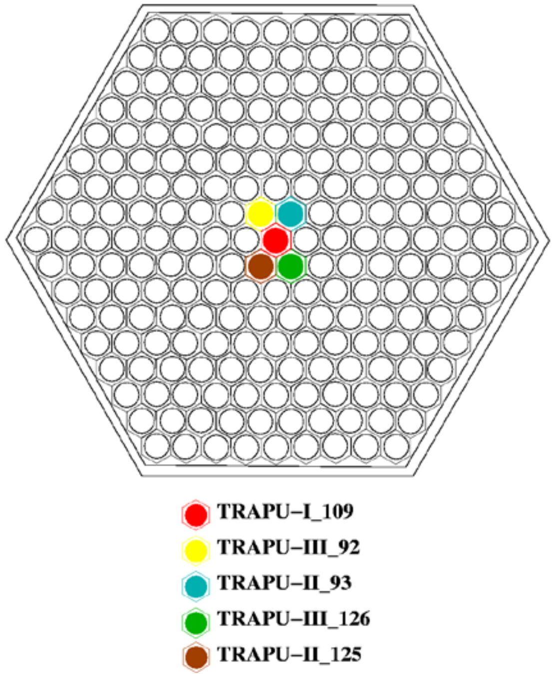

Figure 2. TRAPU Pin Locations in Standard Fuel Assembly of Phenix

Table III. Irradiation Conditions of TRAPU Irradiation Experiment

Irradiation Cycle	Length (days)	Power
Shutdown	140.	
Cycle 10	62.5	P_1 = 379.36MW
Shutdown	103.	
Cycle 11	92.	$P_2 = 1.0075 \times P_1$
Shutdown	50.	
Cycle 12-13	140.6	$P_3 = 1.2965 \times P_1$
Shutdown	48.	
Cycle 14-15	140.1	$P_4 = 1.3462 \times P_1$
Shutdown	105.	

4 Sensitivity Study

According to the specifications of the Nuclear Data Uncertainty Reduction project, sensitivity coefficients have to be generated for the final number density, n_f, and the number density variation, Δn, between EOC and BOC of specific isotopes in the irradiated pins, as indicated in **Table IV**.

Table IV. Sensitivity Analysis for TRAPU Irradiation Experiment

Final Density and Density Variation (EOC – BOC)		
in TRAPU-I of Isotope:	**in TRAPU-II of Isotope:**	**in TRAPU-III of Isotope:**
U234	U234	U234
U235	U235	U235
U236	U236	U236
Np237	Np237	Np237
Pu238	Pu238	Pu238
Pu239	Pu239	Pu239
Pu240	Pu240	Pu240
Pu241	Pu241	Pu241
Pu242	Pu242	Pu242
Am241	Am241	Am241
Am242	Am242	Am242
Am243	Am243	Am243
Cm242	Cm242	Cm242
—	Cm243	Cm243
Cm244	Cm244	Cm244

The major difficulties in performing the required sensitivity study with DPT resulted from the code limitation to a single burn cycle analysis and to the RZ geometry. Thus, the sensitivity study with the DPT code was performed with the following approximations. The RZ model was based only on the Cycle 10 configuration of Phenix, as shown in **Figure 3**. Each of the homogenized regions of core and blankets was represented as a separate depletion zone. As shown in **Figure 3**, the two assemblies CPa 319 and CPa 321 are not described and a region “SAMPLE” was introduced to denote a zone of interest for the sensitivity analysis to be performed: this region has a radial extension of 0.1 cm with 2 cm height and is located at the center of the cylindrical third row of Phenix at the core midplane. In the SAMPLE region the compositions of the irradiated pins TRAPU-I, -II and -III (see **Table V**) are given depending on the case being considered.

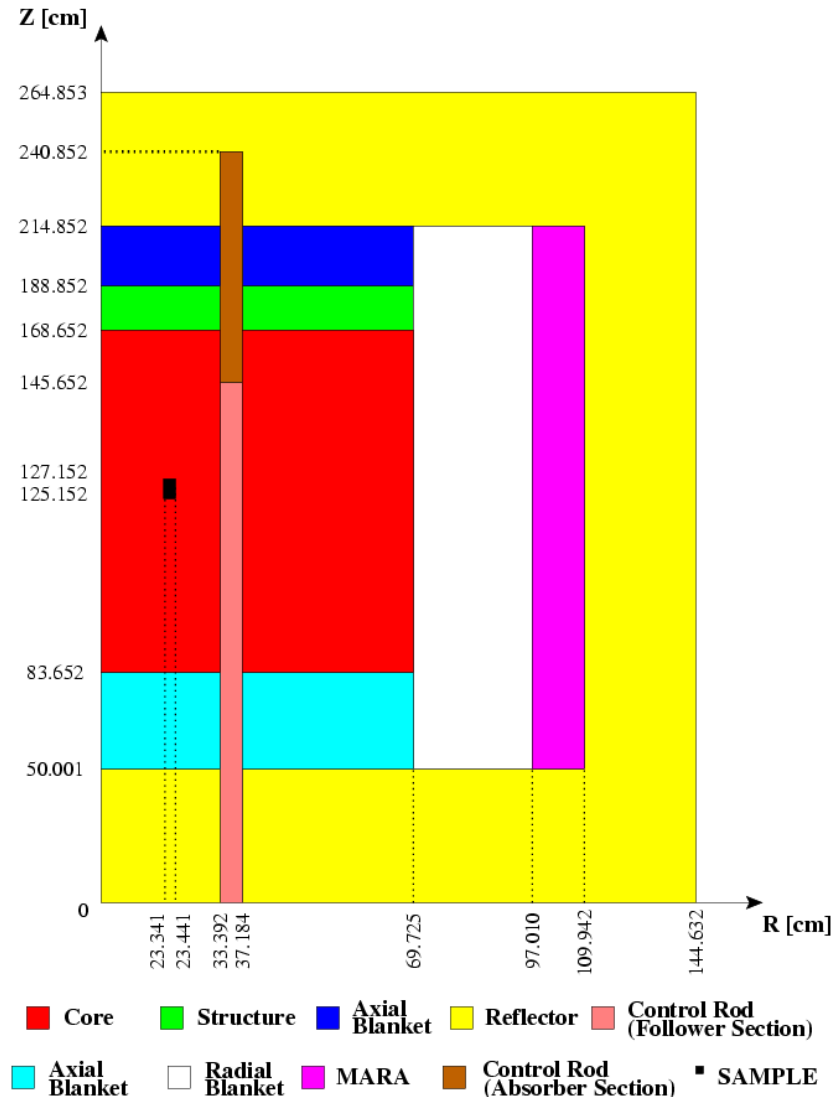

Figure 3. RZ Model of Phenix Configuration Used for the Analysis of the TRAPU Experiment

Table V. Initial Compositions of the TRAPU Pins (Cycle 10)

Isotope	TRAPU-I n. 109	TRAPU-II n. 93	TRAPU-III n. 92
U234	2.224023E-5	3.928308E-5	2.928083E-5
U235	2.692179E-3	2.821711E-3	2.477890E-3
U236	5.560056E-6	9.264877E-6	3.294093E-5
U238	3.706704E-1	3.705951E-1	3.327367E-1
Np237	3.706704E-7	4.076546E-6	1.397494E-5
Pu238	1.097160E-4	6.684979E-4	2.835028E-4
Pu239	6.649457E-2	6.218585E-2	4.348203E-2
Pu240	1.989650E-2	1.615402E-2	6.324027E-2
Pu241	3.620629E-3	6.462354E-3	1.283677E-2
Pu242	6.463272E-4	1.657253E-3	8.169404E-3
Am241	2.433701E-4	1.271701E-3	3.400295E-3

To perform the required sensitivity calculations with the DPT code, a single burn cycle of 458.58 days was derived from the original TRAPU irradiation scheme, excluding the shutdown periods. Using the irradiation history shown in **Table III**, the average power was determined as:

$$\begin{aligned} P_{avg} &= P_1 \times (62.5+92\times 1.0075+140.6\times 1.2965+140.1\times 1.3462)/(62.5+92+140.6+140.1) \\ &= 458.58 \quad \text{MW} \end{aligned}$$

As an example, **Table VI** shows the sensitivities for the final densities of Pu239 in the TRAPU-I pin and the biggest effects are due to the density terms of Pu239 fission and U238 capture. **Table VII** shows the sensitivities for the final densities of U235 in the TRAPU-II pin and the major contributions are due to the density term of U235 fission and the power terms of Pu239 fission. Finally, **Table VIII** shows the sensitivities for the final densities of Am242m in the TRAPU-III pin and the biggest effects are due to the density term of Am241 capture and to a lesser extent to the power terms of Pu239 fission.

Table VI. Sensitivity Coefficients of Pu239 Final Number Density in TRAPU-I

Reaction	BOC-Flux	EOC-Flux	BOC-Power	EOC-Power	Density	Total
U235 fission	- [(a)]	0.001	0.003	0.002	-	0.005
U235 nu	-	0.001	-	-	-	0.001
U238 capture	-0.008	-0.011	0.002	0.002	0.254	0.239
U238 fission	-	0.001	0.009	0.009	-0.001	0.018
U238 nu	0.001	0.002	-	-	-	0.002
U238 elastic	-	-	-	-	-	0.001
U238 inelastic	0.001	0.001	-	-	-	0.002
Pu239 capture	0.001	-	-	-	-0.075	-0.073
Pu239 fission	-0.003	-0.002	0.052	0.051	-0.325	-0.227
Pu239 nu	-0.001	-0.001	-	-	-	-0.002
Pu239 elastic	-	-	-	-	-	-0.001
Pu240 capture	-	-	-	-	-	0.001
Pu240 fission	-	-	0.003	0.004	-	0.007
Pu240 nu	-	-0.001	-	-	-	-0.001
Pu241 fission	-	-0.001	0.003	0.003	-	0.005
Pu241 nu	-	-0.001	-	-	-	-0.001

[(a)] Negligible sensitivities

Table VII. Sensitivity Coefficients of U235 Final Number Density in TRAPU-II

Reaction	BOC-Flux	EOC-Flux	BOC-Power	EOC-Power	Density	Total
U234 capture	-	-	-	-	0.002	0.002
U235 capture	-	-	-	-	-0.105	-0.104
U235 fission	0.002	0.002	0.008	0.007	-0.383	-0.363
U235 nu	0.001	0.002	-	-	-	0.004
U238 capture	0.009	-0.001	0.006	0.006	0.031	0.050
U238 fission	-	0.002	0.029	0.030	-	0.061
U238 nu	0.003	0.005	-	-	-	0.008
U238 elastic	-0.003	-0.003	-	-	-	-0.006
U238 inelastic	-0.026	-0.027	-	-	-	-0.053
Pu239 capture	0.010	0.008	0.001	0.002	-0.005	0.016
Pu239 fission	0.021	0.019	0.169	0.170	-0.025	0.354
Pu239 nu	-0.002	-0.003	-	-	-	-0.006
Pu239 elastic	-0.002	-0.001	-	-	-	-0.003
Pu239 inelastic	-0.003	-0.002	-	-	-	-0.005
Pu240 capture	0.003	0.004	-	-	0.002	0.009
Pu240 fission	-	-0.001	0.011	0.012	-	0.023
Pu240 nu	-	-0.002	-	-	-	-0.002
Pu240 elastic	-0.001	-0.001	-	-	-	-0.001
Pu240 inelastic	-0.001	-0.001	-	-	-	-0.001
Pu241 capture	0.001	0.001	-	-	-	0.001
Pu241 fission	0.001	-	0.011	0.011	-0.002	0.021
Pu241 nu	-0.001	-0.003	-	-	-	-0.004

[a] Negligible sensitivities

Table VIII. Sensitivity Coefficients of Am242m Final Number Density in TRAPU-III

Reaction	BOC-Flux	EOC-Flux	BOC-Power	EOC-Power	Density	Total
U235 capture	-0.001	-	-	-	-	-0.001
U235 fission	-0.003	-0.003	-0.008	-0.007	0.001	-0.019
U235 nu	-0.001	-0.002	-	-	-	-0.003
U238 capture	-0.039	-0.028	-0.006	-0.006	-0.038	-0.116
U238 fission	-	-0.001	-0.027	-0.029	-	-0.057
U238 nu	-0.003	-0.005	-	-	-	-0.008
U238 elastic	0.006	0.005	-	-	-	0.011
U238 inelastic	0.046	0.045	-	-	-	0.091
Pu239 capture	-0.019	-0.017	-0.001	-0.001	0.007	-0.032
Pu239 fission	-0.045	-0.040	-0.161	-0.164	0.032	-0.377
Pu239 nu	0.003	0.003	-	-	-	0.006
Pu239 elastic	0.002	0.002	-	-	-	0.004
Pu239 inelastic	0.005	0.004	-	-	-	0.009
Pu240 capture	-0.008	-0.009	-	-	0.012	-0.006
Pu240 fission	-0.001	-	-0.011	-0.012	-	-0.023
Pu240 nu	-	0.002	-	-	-	0.002
Pu240 elastic	-	-	-	-	-	0.001
Pu240 inelastic	0.001	0.002	-	-	-	0.003
Pu241 capture	-0.001	-0.001	-	-	-0.002	-0.005
Pu241 fission	-0.005	-0.004	-0.010	-0.011	-0.014	-0.045
Pu241 nu	0.001	0.002	-	-	-	0.003
Pu241 inelastic	-	-	-	-	-	0.001
Pu242 capture	-0.001	-0.001	-	-	-	-0.001
Am241 capture	-0.001	-0.001	-	-	0.824	0.822
Am241 fission	-	-	-	-	-0.034	-0.034
Am242m capture	-	-	-	-	-0.034	-0.034
Am242m fission	-	-	-	-	-0.289	-0.289

[a] Negligible sensitivities

5 Conclusions

Within the Nuclear Data Uncertainty Reduction project the ANL tasks are: (1) generation of nuclear data sensitivity coefficients, (2) study of the methods and modeling approximation for integral parameter sensitivities, (3) uncertainty propagation with new covariance data to provide feedback to the nuclear data evaluators, (4) generation of calculational models, calculated values and their uncertainties, and experimental values and their uncertainties, (5) development of data adjustment methodologies, and (6) preliminary application of these cross section adjustment methods.

In the framework of the Nuclear Data Uncertainty Reduction project, one of the major activities assigned to ANL for FY2012 is related to the sensitivity study to be conducted for 82 isotope transmutations in specific samples of the PROFIL-1, PROFIL-2 and TRAPU irradiation experiments. The analysis of these irradiation experiments for the cross section adjustment of the Nuclear Data Uncertainty Reduction project is primarily intended to assess the experimental database on minor actinides made available by the CEA (France) and to validate and improve the current set of minor actinide cross sections and decay constants. The required sensitivity analyses for the PROFIL-1 and PROFIL-2 experiments have been described in a previous report distributed in June 2012. This report documents the now completed studies on the TRAPU experiment. Calculations were performed with the ANL depletion perturbation theory code DPT, which is able to evaluate the impact of the flux distribution change during irradiation on the nuclide density evolution. This impact, though commonly neglected in previous studies based for instance on the ERANOS and NUTS codes, was demonstrated to be non-negligible. The obtained results showed that besides the nuclide density term important sensitivities are also due to the power terms, especially through the Pu239 fission reaction.

The sensitivities obtained for 44 isotope buildups in the TRAPU irradiated pins have been delivered to the other participants in the Nuclear Data Uncertainty Reduction project. A global database is now being created at ANL that is expected to contain in a specific format all sensitivities and C/Es of the several experiments investigated by the different Labs (primarily, ANL and INL) involved in the Nuclear Data Uncertainty Reduction project, as well as detailed information on the codes, models and approximations used for the calculations.

6 References

1. G. Aliberti and R. D. McKnight, Argonne National Laboratory, unpublished information, June 30, 2012.
2. G. Aliberti, W. S. Yang and R. D. McKnight, Argonne National Laboratory, unpublished information, September 30, 2008.
3. G. Aliberti, W. S. Yang and R. D. McKnight, Argonne National Laboratory, unpublished information, September 30, 2009.
4. G. Aliberti, W. S. Yang and R. D. McKnight, Argonne National Laboratory, unpublished information, September 15, 2010.
5. G. Aliberti, W. S. Yang and R. D. McKnight, Argonne National Laboratory, unpublished information, September 30, 2011.
6. A. D'Angelo, F. Cleri, P. Marimbeau, M. Salvatores, and J. P. Grouiller, "Analysis of Sample and Fuel Pin Irradiation in Phenix for Basic Nuclear Data Validation," *Nuclear Science and Engineering*, **105**, 244 (1990).
7. J. Tommasi, "Géométrie et Compositions pour l'Interprétation de l'Expérience PROFIL-2 dans Phenix," Private Communication
8. L. Martin-Deidier, C. Faure, G. Manent, M. Robin, "Résultats des Analyses Effectuées sur les Aiguilles Combustibles de l'Expérience TRAPU," Private Communication
9. G. Palmiotti, G. Aliberti, M. Salvatores, Argonne National Laboratory, unpublished information, September 2004.
10. G. Palmiotti, G. Aliberti, M. Salvatores, J. Tommasi, "Integral Experiments Analysis for Validation and Improvement of Minor Actinide Data for Transmutation Needs," International Conference ND2004, Santa Fe, NM, USA, Sept. 26-Oct. 1, 2004.
11. G. Palmiotti, M. Salvatores, G. Aliberti, H. Hiruta, R. D. McKnight, P. Oblozinsky, W. S. Yang, "A Global Approach to the Physics Validation of Simulation Codes for Future Nuclear Systems," International Conference Physor 2008, Interlaken, Switzerland, September 14-19, 2008.
12. C. H. Lee and W. S. Yang, "Development of Multigroup Cross-section Generation Code MC2-3 for Fast Reactor Analysis," *Proc. of Int. Conf. on Fast Reactors and Related Fuel Cycles (FR09)*, Kyoto, Japan, December 7-11, 2009.
13. M. B. Chadwick *et al*., "ENDF/B-VII.0: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology," *Nucl. Sci. Eng.*, **107**, 2931 (2006).
14. B. J. Toppel, "A User's Guide to the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, Argonne National Laboratory (1983).
15. K. L. Derstine, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," ANL-82-64, Argonne National Laboratory (1984).
16. W. S. Yang and T. J. Downar, "Generalized Perturbation Theory for Constant Power Core Depletion," *Nuclear Science and Engineering*, **99**, February 1988.

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