INL/CON-07-12825
PREPRINT

Importance of the (n,gamma) Cm-247 Evaluation on Neutron Emission in Fast Reactor Fuel Cycle Analysis

2007 ANS/ENS International Meeting

Benoit Forget
Mehdi Asgari
Rodolfo M. Ferrer
Samuel Bays

November 2007

Importance of the (n,gamma) Cm-247 Evaluation on Neutron Emission in Fast Reactor Fuel Cycle Analysis

Benoit Forget, Mehdi Asgari, Rodolfo M. Ferrer and Samuel Bays

Reactor & Nuclear Physics Group
Idaho National Laboratory
Idaho Falls, ID, 83415
Benoit.Forget@inl.gov

INTRODUCTION

As part of the GNEP program, it is envisioned to build a fast reactor for the transmutation of minor actinides. The spent nuclear fuel from the current fleet of light water reactors would be recycled, the current baseline is the UREX+1a process [1], and would act as a feed for the fast reactor. As the fuel is irradiated in a fast reactor a certain quantity of minor actinides would thus build up in the fuel stream creating possible concerns with the neutron emission of these minor actinides for fuel transportation, handling and fabrication.

Past neutronic analyses [2-3] had not tracked minor actinides above Cm-246 in the transmutation chain, because of the small influence on the overall reactor performance and cycle parameters. However, when trying to quantify the neutron emission from the recycled fuel with high minor actinide content, these higher isotopes play an essential role and should be included in the analysis. In this paper, the influence of tracking these minor actinides on the calculated neutron emission is presented. Also presented is the particular influence of choosing a different evaluated cross section data set to represent the minor actinides above Cm-246. The first representation uses the cross-sections provided by MC^2-2 [4] for all isotopes, while the second representation uses infinitely diluted ENDF/BVII.0 [5] cross-sections for Cm-247 to Cf-252 and MC^2-2 for all other isotopes.

METHODOLOGY

The analysis was performed on a 1000MW reference metallic fuel Advanced Burner Reactor (ABR) with a conversion ratio of 0.5 as described by Hoffman [2]. The DIF3D/REBUS [6-7] code package was used for the equilibrium fuel cycle analysis while utilizing the MC^2-2 code as a means to process and self-shield a set of 33-group cross-sections. Additionally, a 33 group cross-section set was generated for the higher minor actinides from the latest ENDF/BVII.0 library using NJOY99 [8] with a typical fast reactor spectrum. The reference case has a transmutation path truncated at Cm-246, while the other two cases included isotopes up to Cf-252. The external feed for all three cases is provided by spent nuclear fuel from a PWR with 51 GWd/MTHM with 5 years of cooling.

RESULTS

Table I compares the neutron emission in the fuel charged in the reactor at the beginning of equilibrium cycle in terms of neutrons per second per metric ton of heavy metal charged in the reactor. The table presents the major contributing isotopes to the total neutron emission as well as the total emission generated by the actinides in the fuel.

Table I: Total Neutron Emission and Principle Contributors (Neutrons/s per MTHM)

	MC^2-2/ Cm-246	**MC^2-2/ Cf-252**	**ENDFVII/ Cf-252**
Cm-242	1.6E9	1.6E9	1.6E9
Cm-244	8.1E10	8.1E10	8.1E10
Cm-246	9.5E9	9.5E9	9.5E9
Cf-250	0	7.5E9	1.7E10
Cf-252	0	1.3E10	3.9E10
Total	9.3E10	1.2E11	1.5E11

An increase of about 30% is observed when including the higher minor actinides in MC^2-2. This increase is attributed mainly to the presence of Cf-250 and Cf-252, which are strong neutron emitters that were previously neglected. Using the ENDFVII library for the higher minor actinides increases the neutron emission by 60% relative to the reference case. This neutron emission increase compared to the MC^2-2 results is due to an increase in the mass of Californium. Table II presents the mass differences at equilibrium in the higher minor actinides between the MC^2-2 and the ENDFVII evaluations in grams per metric ton of heavy metal charged at equilibrium.

Table II: Mass Variations of Higher Minor Actinides

	MC^2-2/ Cf-252	**ENDFVII/ Cf-252**	**Variation (%)**
Cm-247	89	70	-21.3%
Cm-248	44	61	38.6
Bk-249	4.1	5.1	24.4
Cf-250	0.68	1.5	120.6
Cf-251	0.094	0.28	197.9
Cf-252	0.0055	0.017	209.1

When comparing the mass variations between the two different evaluations of higher minor actinides, a mass increase is observed for all isotopes except for the Cm-247. This indicates a possible difference in the (n,gamma) cross-section evaluation of Cm-247.

CURIUM-247

A closer look at the Cm-247 (n,gamma) cross-section in MC^2-2 and ENDF/B-VII.0 indicates that they come from two different and distinct evaluations. Figure 1 illustrates the point-wise cross-section data from the 1976 evaluation that were used in MC^2-2 (ENDF/B-VI and earlier releases) and the 1995 evaluation used in ENDF/B-VII.

As illustrated in figure 1, there is a significant difference in the (n,gamma) cross-section in the fast energy range between both evaluations, especially when one considers that the plot is in a log-log scale. Figure 2 takes a closer look at the fast energy range by focusing in the 100 eV to 10 MeV range for the 33 group cross-section set.

To analyze the effect of the MC^2-2 calculation on the Cm-247 isotope the ENDF/B-VI (1976 evaluation) point-wise data was processed with a reference fast spectrum to provide a 33 group data set just like was done for the ENDF/B-VII (1995 evaluation) data set. This cross-section set agreed very well with the MC^2-2 data, thus indicating that the energy self-shielding provided by MC^2-2 has minimal effect on this particular isotope for this specific case and that the infinitely dilute approximation is somewhat justified. In addition to the cross-section plot, the (n,gamma) reaction rates for both set of data was also plotted in figure 2. As illustrated, the reaction rate with the ENDF/B-VII data is much larger, thus indicating a reduction of Cm-247 and a movement up the minor actinide chain.

CONCLUSION

The addition of the higher minor actinides (above Cm-246) in the transmutation chain has little to no impact on the neutronics and cycle performance parameters of a fast reactor with minor actinide recycling. However, a significant increase in neutron emission is identified. The different evaluations of the Cm-247 isotope also play a major role in quantifying the neutron emission from the fuel. The 1976 evaluation that was used in ENDF/B-VI and earlier releases presents a much lower (n,gamma) cross-section value for Cm-247 than the 1995 evaluation. This change in cross-section impacts greatly the movement up the actinide chain, thus impacting the neutron emission from Californium isotopes.

Cross-section uncertainty in the higher minor actinides has a significant impact on the uncertainty of neutron emission estimates. Future sensitivity studies should include these minor actinides and find reasonable ways to estimate a covariance matrix for this data.

REFERENCES

1- G.F.Vandegrift *et al*, “Design and demonstration of the UREX+ Process Using Spent Nuclear Fuel,” Advances for Future Nuclear Fuel Cycles, Nimes, France, June 21-24 (2004).
2- E.A.Hoffman, W.S.Yang and R.N.Hill, “Preliminary Core Design Studies for the Advanced Burner Reactor over a Wide Range of Conversion Ratios,” Argonne National Laboratory, ANL-AFCI-177 (2006).
3- R.M.Ferrer, M.Asgari, S.Bays and B.Forget “Computational Neutronics Methods and Transmutation Performance Analyses for Fast Reactors,” Idaho National Laboratory, INL-EXT-07-12466 (2007).
4- H.Henryson II. B.J.Toppel, and C.G.Stenberg, “MC^2-2: A code to Calculate Fast Neutron Spectra and Multi-Group Cross-Sections,” Argonne National Laboratory, ANL-8144 (1976).
5- M.B.Chadwick, “ENDF/B-VII.0: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology,” Nuclear Data Sheets, **107**, 12 (2006).
6- K.L.Derstine, “DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite Difference Diffusion Theory Problems,” Argonne National Laboratory, ANL-82-64 (1984).
7- B.J.Toppel, “A User’s Guide to the REBUS-3 Fuel Cycle Analysis Capability,” Argonne National Laboratory (1983).
8- R.E.MacFarlane and D.W.Muir, “The NJOY nuclear data processing system, version 91,” Tech. Rep. LA-12740-M, Los Alamos National Lab, NM (1994).

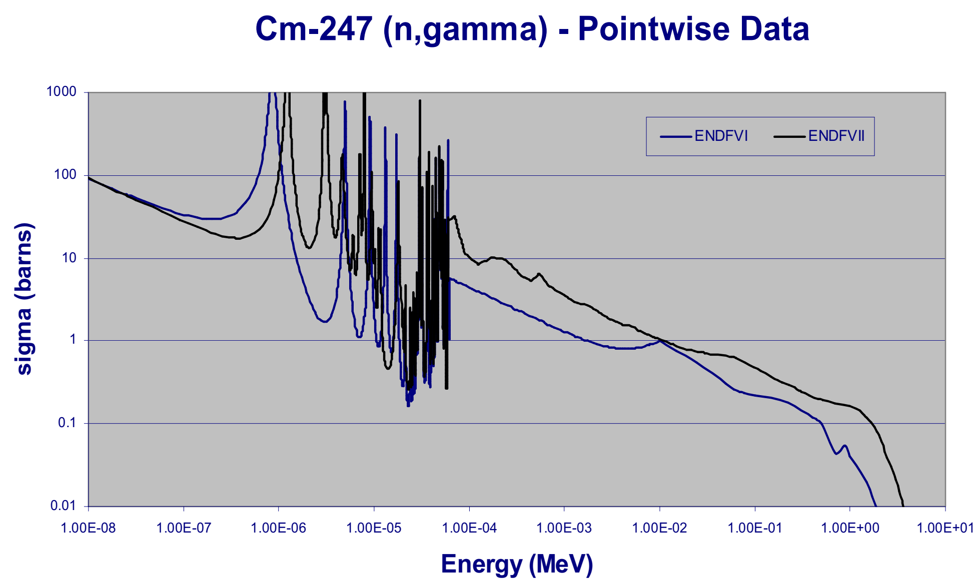

Figure 1: Point-wise Cross-section Plots of the (n,gamma) Cm-247 Reaction

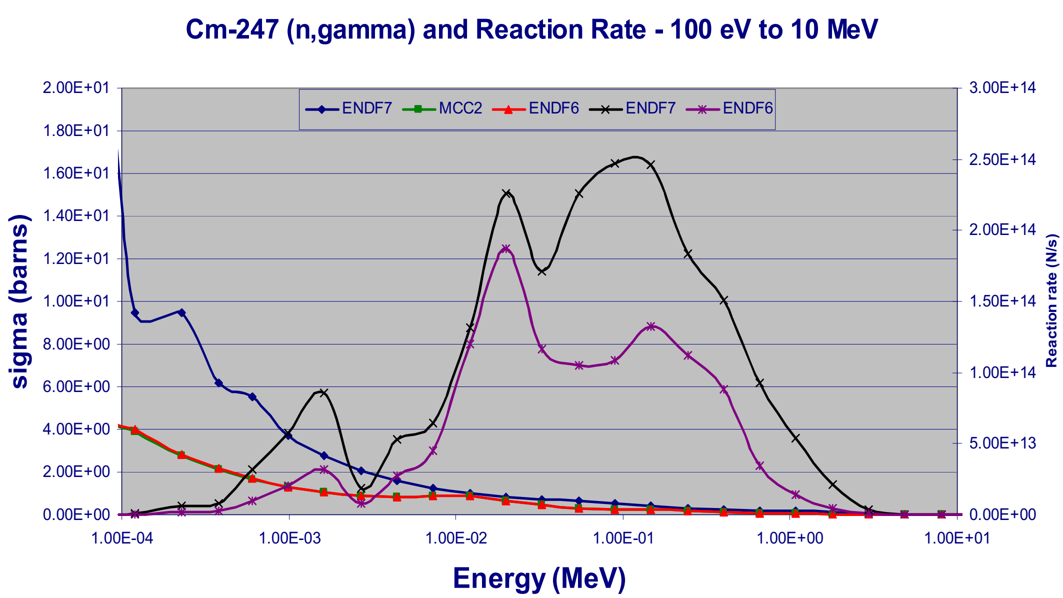

Figure 2: 33 Group Cross-section Data and Reaction Rates in Fast Energy Range