IMPACT OF RECENT ENDF NUCLEAR DATA ON BURNUP CREDIT CRITICALITY SAFETY ANALYSES*

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ABSTRACT

Criticality safety analyses are performed to ensure compliance with regulatory and safe operational requirements in storing and transporting spent nuclear fuel (SNF). SNF at discharge has a reduced reactivity compared to that of fresh fuel because of burnup. The consideration of the reduced reactivity of SNF in criticality safety studies is referred to as burnup credit. In 2012, NUREG/CR-7109 was created to provide users with guidelines regarding burnup credit studies. The studies performed in NUREG/CR-7109 utilized the ENDF/B-VII.0 cross section and the 44-group covariance data for reaction probabilities. Since the publishing of NUREG/CR-7109 in 2012, continuous efforts have been made to improve the cross section and covariance data. The ENDF library saw two major releases, ENDF/B-VII.1 in 2011 and ENDF/B-VIII.0 in 2018. In addition, a new 252-group covariance data set has been released. It is important to reanalyze the burnup studies with the new cross section and covariance data set to quantify their effect. A nominal pressurized water reactor assembly was studied at two assembly average burnups and two initial U-235 fuel enrichments. The assemblies were placed in a generic burnup credit cask GBC-32. The ENDF/B-VIII.0 data library resulted in lower nuclear data uncertainties than the ENDF/B-VII.1 data library. The differences in the nuclear data–induced uncertainties resulting from the use of ENDF/B-VII.1 and ENDF/B-VIII.0 data libraries for most nuclides was 2% or lower.

KEYWORDS

Burnup credit, sensitivity and uncertainty, nuclear data uncertainties, nuclear covariance data

1. INTRODUCTION

Criticality safety analyses must be performed to ensure compliance in safely storing and transporting spent nuclear fuel (SNF). Because of burnup, SNF has a reduced reactivity compared to that of fresh fuel. The consideration of the reduced reactivity of SNF in criticality safety studies is referred to as *burnup credit*. In 2012, the Nuclear Regulatory Commission Contractor Report NUREG/CR-7109 [1] was developed to provide users with guidelines for burnup credit studies. The research documented in NUREG/CR-7109 was performed using the SCALE 6.1 code package [2] with the ENDF/B-VII.0 cross section libraries and the 44-group covariance data for reaction probabilities [3]. The conclusions of NUREG/CR-7109 showed that sufficient benchmark experiments were available to perform burnup credit criticality safety analyses with SCALE and the ENDF/B-VII.0 nuclear data. The study also

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showed that the uncertainty in k_{eff} caused by nuclear data uncertainties conservatively bounds the biases associated with nuclear data errors.

Efforts to improve the cross section and covariance data have persisted, and since the completion of NUREG/CR-7109, the ENDF library has seen two major releases: ENDF/B-VII.1 in 2011 [4] and ENDF/B-VIII.0 in 2018 [5]. In addition, a new 252-group covariance data set has been released. Therefore, it is essential to analyze the burnup studies with the new cross section and covariance data set to quantify their effect.

This work presented herein describes the effects of using the updated nuclear data libraries and covariance data on k_{eff} and its nuclear data–induced uncertainties. The uncertainties in k_{eff} from major actinide, minor actinide, and fission products are reported. Pressurized water reactor (PWR) fuel assemblies of varying enrichments and burnup were placed in a spent fuel dry cask to analyze the effect of using these libraries.

2. MODEL DESCRIPTION

A generic burnup credit cask model GBC-32 was used for this analysis. The cask has a capacity of 32 PWR fuel assemblies. The GBC-32 model includes 32 square storage cells. The cells have an inner dimension of 22 cm, and they are surrounded by stainless-steel walls and Boral® panels. The cask was loaded with Westinghouse 17×17 optimized fuel assemblies (OFAs). The fuel assemblies were centered in the cask storage cells. The cask and fuel are described in detail in NUREG/CR-6747 [6] and NUREG/CR-7109 [1]. Figure 1 shows a cutaway view of the cask and fuel.

Two assembly average burnups, 40 and 60 GWd/MTU, and two initial U-235 fuel enrichments, 4 and 5 wt %, were considered in this study. The SCALE TRITON "t-depl" sequence and the ORIGEN program were used for lattice physics and depletion, fuel modeling, and generation of the fuel inventories for the various burnup, enrichment, and depletion conditions. A 5-year cooling period was assumed for the fuel loaded in the cask.

The SCALE CSAS5 sequence was used for the criticality calculations to calculate the k_{eff} values and their uncertainties for the different models. The TSUNAMI-3D sequence was used to determine the sensitivity of the calculated k_{eff} values to the nuclear data. The sensitivities are reported as a function of the nuclide and nuclear reaction. Central difference direct perturbations were also performed to verify TSUNAMI-3D sensitivity results. Finally, the TSUNAMI-IP sequence was used to generate the nuclear data–induced uncertainties.

The SCALE 6.3 code package [7] with ENDF/B-VII.1 and ENDF/B-VIII.0 cross section libraries and covariance data were used in this study. Both the ENDF/B-VII.1 and ENDF/B-VIII.0 libraries include 252 neutron energy groups and 56 covariance groups.

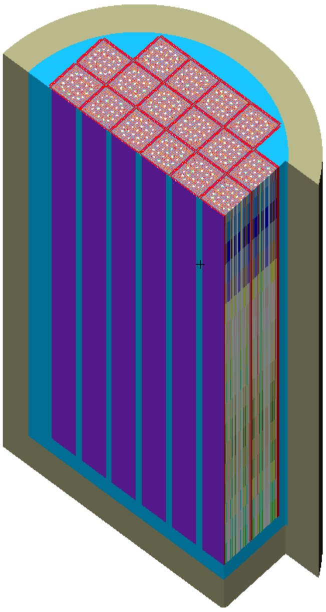

Figure 1: Cutaway of the GBC-32 cask and PWR fuel assemblies

3. RESULTS AND DISCUSSION

Table I summarizes the results obtained for the different fuel enrichments (in wt % U-235) and burnup (in GWd/MTU) using the ENDF/B-VII.1 and ENDF/B-VIII.0 cross section data libraries. The uncertainty in k_{eff} is approximately 10 per cent mille (pcm). The maximum difference in k_{eff} between ENDF/B-VII.1 and ENDF/B-VIII.0 is 0.55%. Table I also shows the total nuclear data uncertainty for the spent fuel in GBC-32 as well as individual nuclide contribution. The majority of the total nuclear data–induced uncertainty results from the major actinides. The difference in these uncertainties resulting from the use of different nuclear data libraries was further analyzed.

Table I. Summary results for different fuel enrichments and burnup

Enrichment	4				5			
Burnup	40		60		40		60	
Library	**E7.1**	**E8**	**E7.1**	**E8**	**E7.1**	**E8**	**E7.1**	**E8**
k_{eff}	0.8978	0.89468	0.8228	0.81833	0.95815	0.95556	0.88303	0.87937
Nuclear Data Uncertainty (Δk)								
Total	**3.91E-03**	**3.90E-03**	**3.79E-03**	**3.78E-03**	**4.00E-03**	**3.99E-03**	**3.82E-03**	**3.82E-03**
Major Actinides	**3.65E-03**	**3.63E-03**	**3.52E-03**	**3.50E-03**	**3.74E-03**	**3.73E-03**	**3.54E-03**	**3.53E-03**
U-234	7.81E-07	7.61E-07	1.33E-06	1.29E-06	8.66E-07	8.43E-07	1.40E-06	1.36E-06
U-235	2.02E-03	2.03E-03	1.49E-03	1.50E-03	2.42E-03	2.44E-03	1.90E-03	1.91E-03
U-238	2.00E-03	1.99E-03	1.86E-03	1.85E-03	1.97E-03	1.95E-03	1.86E-03	1.85E-03
Pu-238	8.87E-05	8.71E-05	1.81E-04	1.78E-04	7.81E-05	7.66E-05	1.59E-04	1.57E-04
Pu-239	2.16E-03	2.14E-03	2.36E-03	2.34E-03	1.96E-03	1.94E-03	2.16E-03	2.14E-03
Pu-240	2.08E-04	2.04E-04	2.53E-04	2.48E-04	1.87E-04	1.83E-04	2.30E-04	2.26E-04
Pu-241	5.58E-04	5.41E-04	7.97E-04	7.80E-04	4.48E-04	4.34E-04	6.50E-04	6.36E-04
Pu-242	1.63E-04	1.51E-04	2.57E-04	2.35E-04	1.42E-04	1.31E-04	2.26E-04	2.08E-04
Am-241	4.40E-04	4.28E-04	5.81E-04	5.66E-04	3.96E-04	3.85E-04	5.36E-04	5.23E-04
Minor Actinides	**2.11E-04**	**2.10E-04**	**2.54E-04**	**2.51E-04**	**2.34E-04**	**2.33E-04**	**2.77E-04**	**2.75E-04**
Am-243	1.77E-05	1.69E-05	3.92E-05	3.72E-05	1.42E-05	1.35E-05	3.15E-05	3.01E-05
Np-237	1.37E-04	1.35E-04	1.86E-04	1.84E-04	1.43E-04	1.41E-04	1.94E-04	1.92E-04
U-236	1.60E-04	1.60E-04	1.68E-04	1.67E-04	1.85E-04	1.84E-04	1.95E-04	1.94E-04
Fission Products	**5.46E-04**	**5.44E-04**	**6.35E-04**	**6.31E-04**	**5.66E-04**	**5.64E-04**	**6.50E-04**	**6.47E-04**
Mo-95	4.18E-05	4.16E-05	4.81E-05	4.75E-05	4.58E-05	4.56E-05	5.21E-05	5.18E-05
Tc-99	8.61E-05	8.49E-05	1.06E-04	1.05E-04	8.78E-05	8.67E-05	1.07E-04	1.06E-04
Ru-101	7.28E-05	7.22E-05	8.96E-05	8.85E-05	7.91E-05	7.86E-05	9.55E-05	9.47E-05
Rh-103	1.89E-04	1.87E-04	2.10E-04	2.07E-04	2.02E-04	1.99E-04	2.22E-04	2.20E-04
Ag-109	2.43E-05	2.35E-05	3.34E-05	3.22E-05	2.28E-05	2.20E-05	3.10E-05	3.00E-05
Cs-133	1.52E-04	1.50E-04	1.74E-04	1.71E-04	1.64E-04	1.62E-04	1.86E-04	1.83E-04
Sm-147	4.83E-05	4.94E-05	5.31E-05	5.38E-05	5.34E-05	5.45E-05	5.87E-05	5.99E-05
Sm-149	1.91E-04	1.89E-04	1.77E-04	1.74E-04	2.02E-04	2.00E-04	1.86E-04	1.84E-04
Sm-150	4.54E-05	4.51E-05	5.93E-05	5.86E-05	4.51E-05	4.48E-05	5.78E-05	5.73E-05
Sm-151	1.18E-04	1.17E-04	1.26E-04	1.24E-04	1.22E-04	1.21E-04	1.29E-04	1.28E-04
Sm-152	5.80E-05	5.70E-05	6.47E-05	6.34E-05	6.08E-05	5.95E-05	6.72E-05	6.60E-05
Nd-143	3.39E-04	3.37E-04	3.92E-04	3.89E-04	3.49E-04	3.47E-04	4.05E-04	4.03E-04
Nd-145	1.52E-04	1.57E-04	1.78E-04	1.82E-04	1.68E-04	1.73E-04	1.94E-04	1.99E-04
Eu-151	2.17E-06	2.16E-06	2.28E-06	2.25E-06	2.35E-06	2.34E-06	2.44E-06	2.42E-06
Eu-153	8.41E-05	8.30E-05	1.14E-04	1.12E-04	8.26E-05	8.15E-05	1.10E-04	1.09E-04
Gd-155	1.47E-04	1.47E-04	2.25E-04	2.25E-04	1.25E-04	1.25E-04	1.93E-04	1.93E-04

Figures 2, 3, and 4 show the differences in nuclear data uncertainties resulting from ENDF/B-VII.1 and ENDF/B-VIII.0 data libraries for major actinides, minor actinides, and fission products, respectively. The ENDF/B-VII.1 results are the comparison base for the percentage difference calculations. The positive percentage difference shows that ENDF/B-VII.1 results in a higher data-induced uncertainty than ENDF/B-VIII.0 for most examined nuclides. In addition, most of the nuclides are within a 2% difference. However, some nuclides, such as Pu-242, Am-243, and Ag-109, exceed a 3% difference. Although Pu-242 has a small nuclear data uncertainty, investigating this result further would be worthwhile because of the 8% difference in Figure 2. Figure 5 shows the total and capture neutron cross section sensitivities using the ENDF/B-VII.1 and ENDF/B-VIII.0 data libraries. The sensitivity is the ratio of the relative difference of k_{eff} to the relative difference of the cross section.

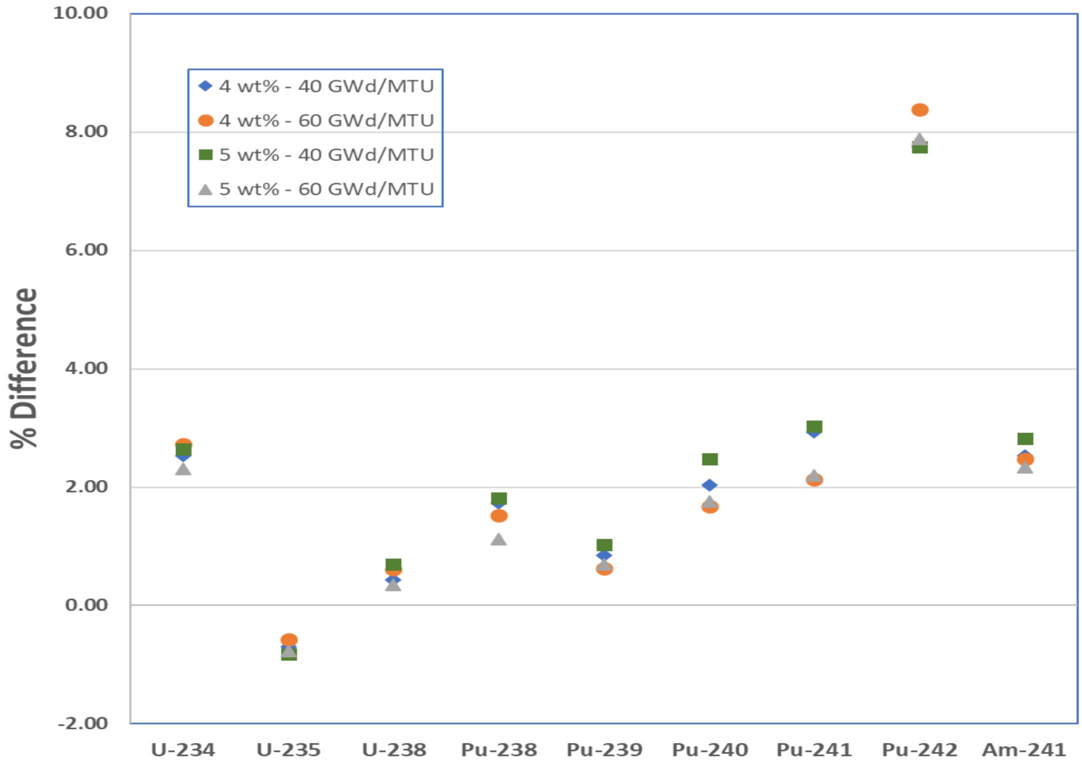

Figure 2: Nuclear data library differences for major actinides

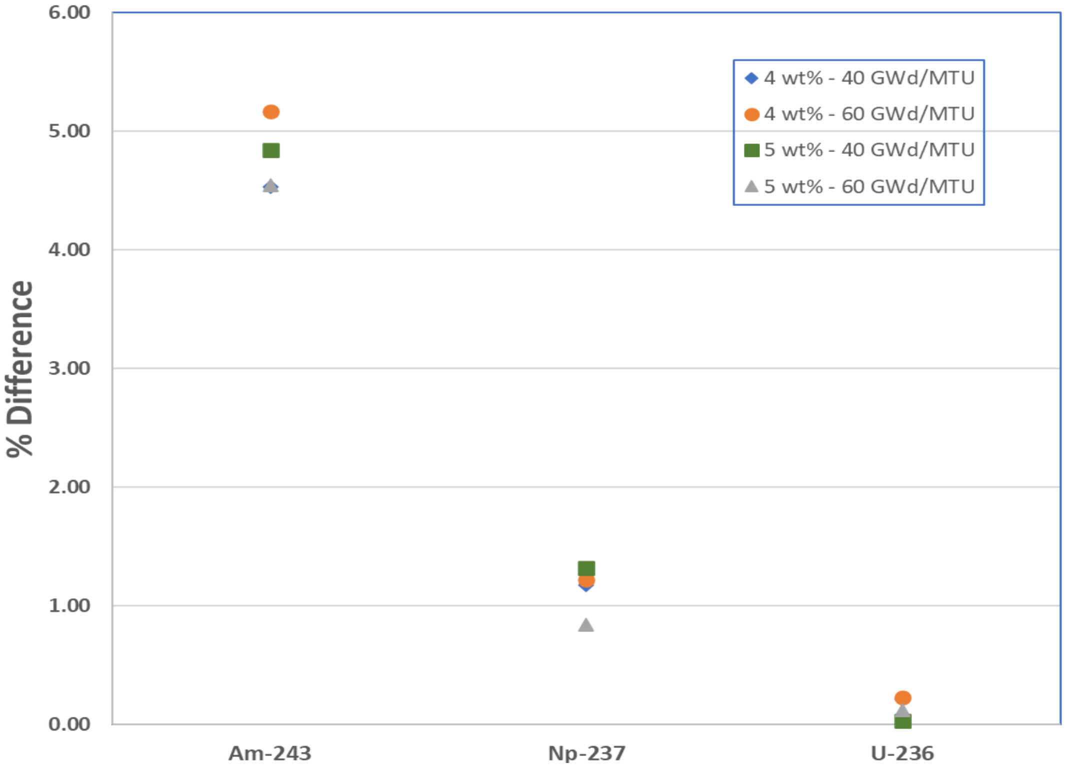

Figure 3: Nuclear data library differences for minor actinides

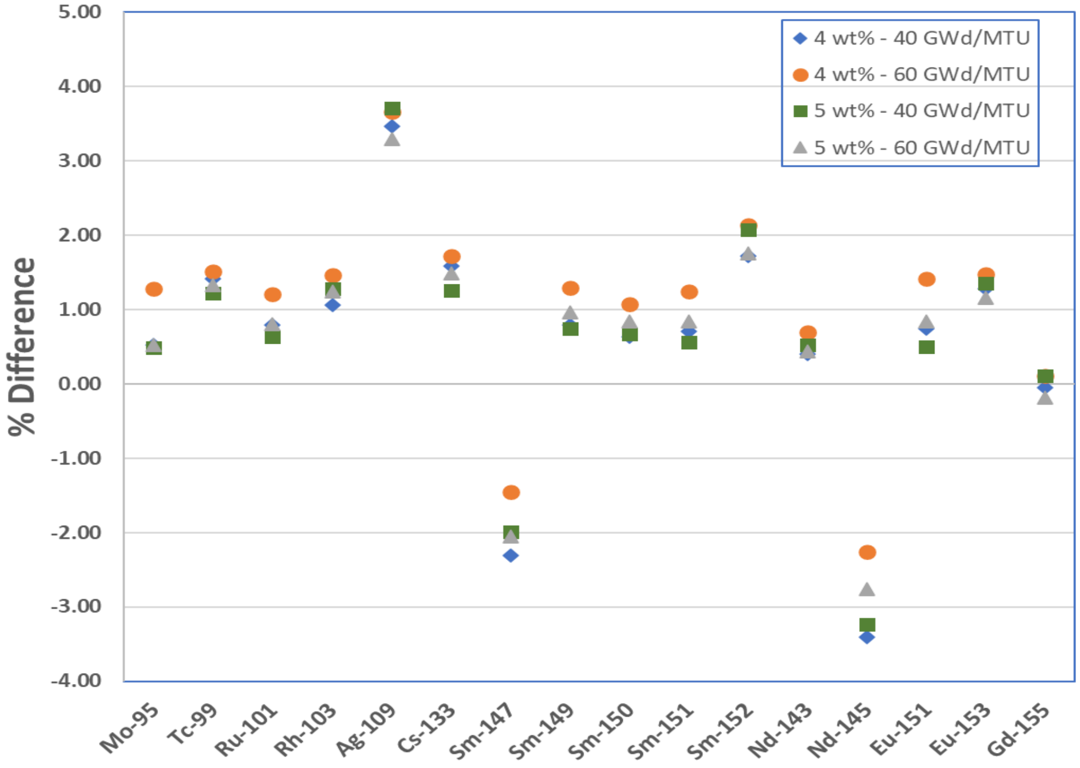

Figure 4: Nuclear data library differences for fission products

As seen in Figure 5, the main difference in the sensitivities for the different cross section libraries is at the thermal neutron energies. Plutonium-242 is mainly produced from a series of neutron captures in the Pu isotopes. The differences in the Pu-242 sensitivities between ENDF/B-VII.1 and ENDF/B-VIII.0 in Figure 5 are mostly caused by the minor differences in the capture cross-section in the data libraries, especially for Pu-240, which resulted in a difference in the Pu-242 concentration

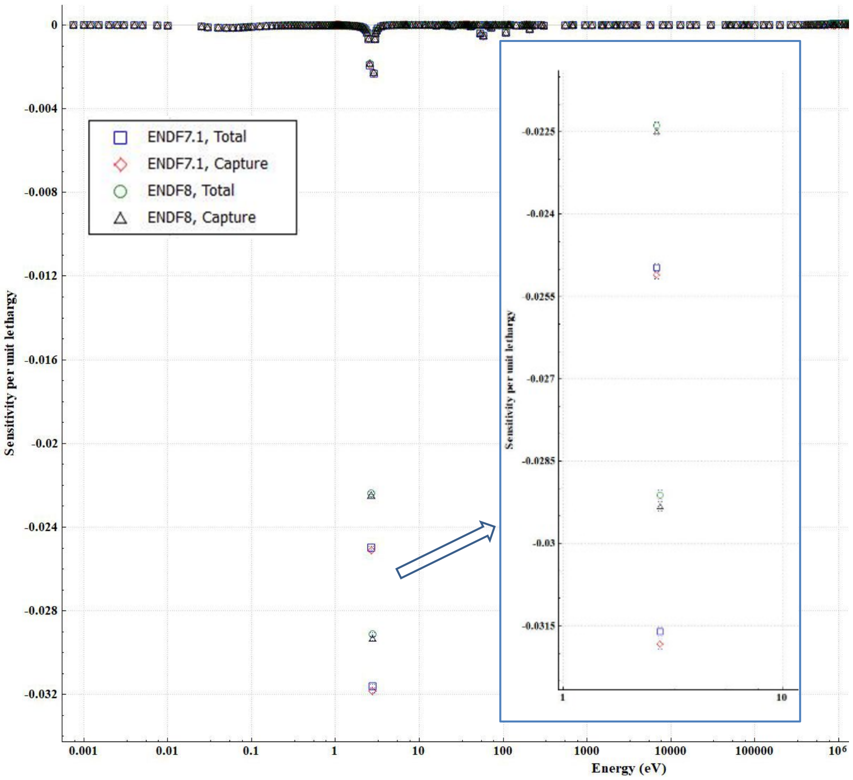

Figure 5: Plutonium-242 neutron cross section sensitivities for neutron total and radiative capture reactions

4. CONCLUSION

In this work, the effects of using the ENDF/B-VII.1 and ENDF/B-VIII.0 nuclear data libraries on k_{eff} were evaluated, as well as the nuclear data–induced uncertainties. A nominal PWR assembly was studied at two assembly average burnups, 40 and 60 GWd/MTU, and two initial U-235 fuel enrichments, 4 and 5 wt %. The assemblies were placed in a generic burnup credit cask GBC-32. The maximum difference in k_{eff} between the data libraries was 0.55%. The differences in the nuclear data–induced uncertainties resulting from ENDF/B-VII.1 and ENDF/B-VIII.0 data libraries for major actinides, minor actinides, and fission products were around 2%, and the use of ENDF/B-VIII.0 data libraries resulted in lower data-induced uncertainties.

ACKNOWLEDGMENTS

This work was performed under contract with the US Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research.

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