Effect of Decay Time on Criticality Safety Analyses for High-Burnup and Extended Enrichment Fuels

A. M. Shaw*, W. A. Metwally*, M. N. Dupont*, W. J. Marshall*, C. Celik*, V. Karriem*, A. Lang*, and K. L. Fassino*

** Oak Ridge National Laboratory, 1 Bethel Valley Road, Oak Ridge, Tennessee, 37830, USA*
shawam@ornl.gov

[leave space for DOI, which will be inserted by ANS]

INTRODUCTION

The determination of system upper subcritical limits (USLs) for safe fissile material transport and storage depends heavily on the validation of such systems, resulting in the necessary determination of biases and bias uncertainties. These values are used to quantify the calculational misprediction of a computational code modeling a real system—as well as to quantify the uncertainty in said value as a result of the suite of applicable critical benchmarks and the methodology employed in determining and accounting for those benchmarks.

The determination of a USL for a spent fuel system is further complicated relative to a fresh fuel system, requiring additional detail regarding the burnup history and appropriate operating parameters. The variation in attributes such as burnable absorber exposure, soluble boron content, and component temperatures result in known spectral and associated reactivity effects [1]. The impact of such variations has also been demonstrated on the determination of bias and bias uncertainties, that is, the sensitivity of bias and bias uncertainty to operating parameters [2]. Generally, the variation in parameters such as fuel temperature, specific power, etc. causes indistinguishable changes in the bias and bias uncertainty [2].

With increased effort and research into the use of burnup credit methods at higher enrichments and burnups, both the underlying changes to spectral and reactivity effects [3] and the variation in bias and bias uncertainty [4] have been investigated with respect to increased enrichment and burnup of pressurized water reactor (PWR) fuel. One parameter with a noted, if minor, variation is the cooling time of the fuel following discharge. A slight trend on the ^{239}Pu inventory was observed, a parameter obviously relevant for increased enrichment and burnup [2]. Therefore, as part of a larger study into bias and bias uncertainty trends with increased enrichment and burnup fuels [4], a sensitivity study was performed to detail the cooling time behavior.

MODEL DESCRIPTION

A generic burnup credit cask model GBC-32 with a capacity of 32 PWR fuel assemblies was used in this work [5]. 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). Fig. *1* shows a cutaway view of the cask and fuel.

The SCALE 6.3 code package [6] with the 252 group ENDF/B-VII.1 [7] and ENDF/B-VIII.0 [8] cross section libraries was used in this study.

Two assembly average burnups, 60 and 80 GWd/MTU, and two initial ^{235}U fuel enrichments, 5 and 7 wt %, were considered in this study. The SCALE TRITON sequence with KENO was used for lattice physics and depletion, fuel modeling, and generation of the fuel inventories for the various burnup, enrichment, and depletion conditions. A cooling period of up to 40 years 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 SCALE VADER tool was used to calculate the bias and bias uncertainties trending on energy of the average lethargy of neutrons causing fission (EALF) with the methodology as determined in NUREG/CR-6698 [9]. The validation suites vary by burnup and enrichment pairing, as determined by similarity assessment, and are noted in [4].

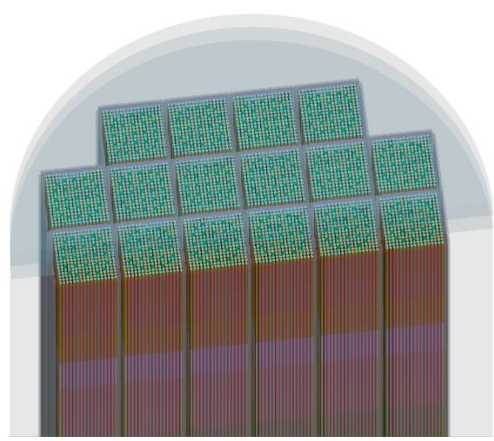

Fig. 1. GBC-32 cask model with fuel assemblies.

TABLE I. Bias (β) and Bias Uncertainties (σ_β) with Variable Conditions

Decay time	Initial enrichment (wt% ^{235}U)	Final burnup (GWd/MTU)	Library	β	σ_β	β + σ_β (pcm)	Simple average: Final enrichment (wt% ^{235}U)	Simple average: Pu/(Pu+U) (wt%)	Fission density-weighted: Final enrichment (wt% ^{235}U)	Fission density-weighted: Pu/(Pu+U) (wt%)
0 days	5	60	E7.1	141	884	1025	1.051	1.563	1.779	1.293
			E8.0	95	917	1012	1.016	1.557	1.763	1.282
	7	60	E7.1	184	925	1109	2.227	1.597	3.157	1.324
			E8.0	82	945	1027	2.182	1.591	3.134	1.315
		80	E7.1	177	943	1120	1.365	1.860	2.544	1.507
			E8.0	79	945	1024	1.318	1.853	2.521	1.496
1 year	5	60	E7.1	141	889	1030	1.051	1.572	1.814	1.284
			E8.0	91	922	1013	1.016	1.565	1.796	1.275
	7	60	E7.1	185	930	1115	2.227	1.602	3.206	1.312
			E8.0	80	949	1029	2.182	1.597	3.184	1.303
		80	E7.1	178	948	1126	1.365	1.868	2.589	1.496
			E8.0	76	950	1026	1.318	1.860	2.565	1.486
5 years	5	60	E7.1	142	893	1035	1.051	1.533	1.873	1.234
			E8.0	89	926	1015	1.016	1.527	1.855	1.225
	7	60	E7.1	185	934	1119	2.227	1.562	3.281	1.262
			E8.0	78	953	1031	2.182	1.557	3.260	1.253
		80	E7.1	179	954	1133	1.365	1.821	2.666	1.439
			E8.0	73	955	1028	1.319	1.814	2.643	1.429
10 years	5	60	E7.1	142	896	1038	1.051	1.492	1.921	1.188
			E8.0	87	929	1016	1.016	1.487	1.904	1.179
	7	60	E7.1	186	936	1122	2.227	1.521	3.344	1.216
			E8.0	76	955	1031	2.182	1.516	3.321	1.209
		80	E7.1	180	958	1138	1.365	1.772	2.728	1.386
			E8.0	71	958	1029	1.319	1.766	2.704	1.377
20 years	5	60	E7.1	142	899	1041	1.052	1.435	1.974	1.130
			E8.0	85	932	1017	1.016	1.430	1.954	1.123
	7	60	E7.1	186	938	1124	2.227	1.462	3.408	1.160
			E8.0	76	957	1033	2.182	1.458	3.387	1.154
		80	E7.1	180	961	1141	1.366	1.702	2.796	1.319
			E8.0	69	962	1031	1.319	1.697	2.772	1.311
40 years	5	60	E7.1	142	900	1042	1.052	1.374	2.016	1.076
			E8.0	84	933	1017	1.016	1.370	1.997	1.070
	7	60	E7.1	186	939	1125	2.228	1.401	3.461	1.108
			E8.0	75	958	1033	2.182	1.397	3.438	1.102
		80	E7.1	181	963	1144	1.366	1.627	2.850	1.256
			E8.0	68	963	1031	1.319	1.622	2.825	1.249

RESULTS

Decay times were investigated at 0, 1, 5, 10, 20, and 40 years following depletion. This reflects the values used in NUREG/CR-7109 with the addition of a 10 year point [2]. TABLE I presents the variations in bias (β), bias uncertainty (σ_β), and the resulting sum. All data are presented in pcm. Results are presented for variations in decay time, initial enrichment, final burnup, and nuclear data library. In addition, the spent fuel compositions post-irradiation and cooldown are summarized with simple and fission-density-weighted averages of the ^{235}U enrichment and Pu/U+Pu weight ratio. This provides an indication of the system's major fissile species in terms of whether a relation is observed between fissile contents and bias and bias uncertainty.

For instance, Fig. *2* demonstrates the variation in bias with the Pu/U+Pu ratio. As a function of cooling time, no significant variation occurs on the EALF-trended biases. Values presented were generated using ENDF/B-VII.1 data, grouped by initial enrichment (wt%) and final burnup (BU), with groupings labeled as "wt%_BU". No variation in the bias with respect to EALF is noticeable in Fig. *3*.

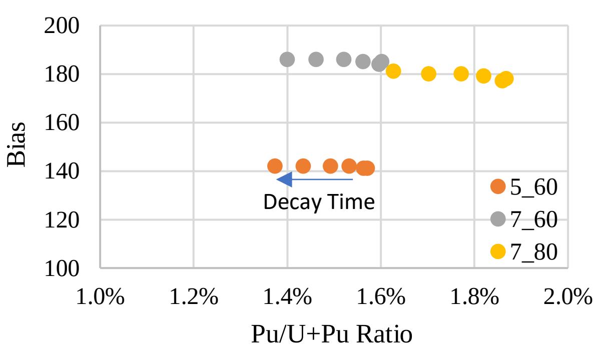

Fig. 2. Variation of bias with Pu/U-Pu ratio.

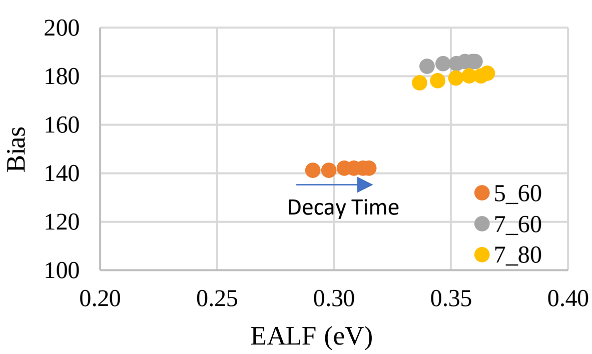

Fig. 3. Variation of bias with EALF.

Fig. *4* demonstrates the variation in bias uncertainty with the Pu/U-Pu ratio. A minor variation occurs, with ~20 pcm difference between the 0 year and 40 year cooling times. Values were generated with ENDF/B-VII.1 data, grouped by initial enrichment (wt%) and final burnup (BU), with groupings labeled as "wt%_BU". Further investigation indicated that the resulting variation in σ_β is more directly related to the spectral hardening due to the increased decay period, as shown in Fig. *5*.

Investigating the suite of experiments for each grouping, this is sensible and expected: there are few critical experiments characteristic of spent fuel, and there are fewer yet with variations in decay times. With a small suite of applicable experiments for the initial spent fuel, further decay pushes the trend parameter, EALF, higher. With this comes increased statistical uncertainty because there are fewer experiments at similar EALFs, which increases σ_β with increased distance from the centroid. Also as expected, this variation between decay times is very minor, on the order of the stochastic uncertainty in the underlying eigenvalue calculation. Changes in fuel are minor after all, with the same design and composition, apart from the slight increase in enrichment and burnup.

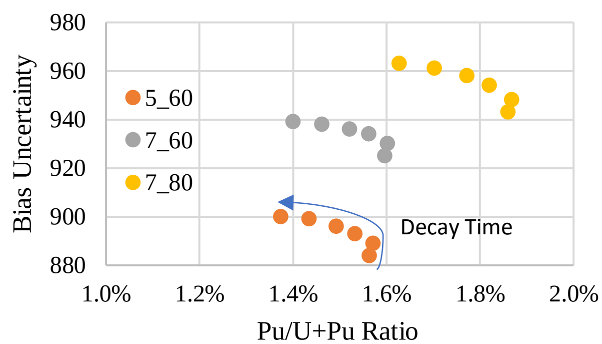

Fig. 4. Variation of bias uncertainty with Pu/U-Pu ratio.

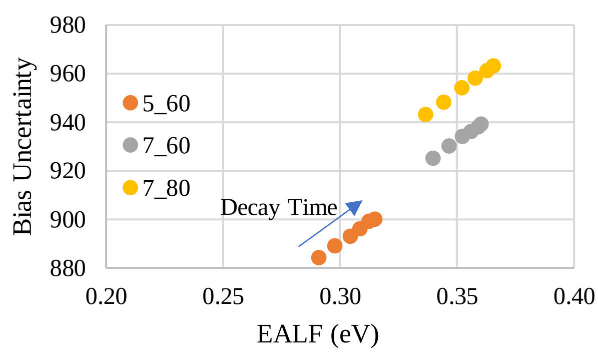

Fig. 5. Variation of bias uncertainty with EALF.

Bias uncertainties for ENDF/B-VIII.0 behave quite similarly to ENDF/B-VII.1 and are not pictured. However, Figure 6 and 7 detail the variation in bias. While slight (7-11 pcm) and well outweighed by the bias uncertainty, there is a steady and consistent downward movement with increasing EALF.

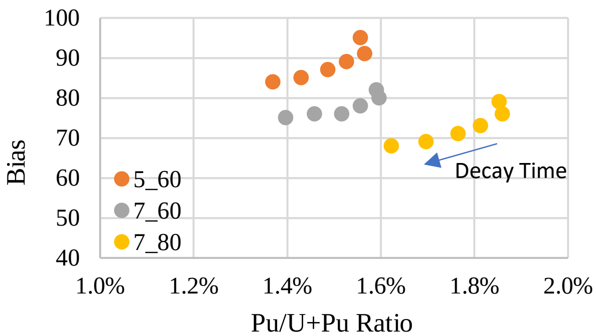

Fig. 6. Variation of bias with Pu/U-Pu ratio: ENDF/B-VIII.0.

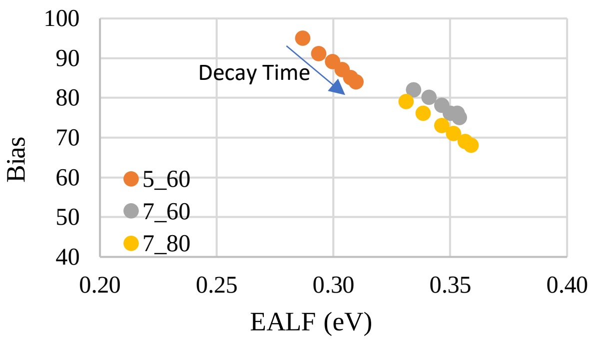

Fig. 7. Variation of bias with EALF: ENDF/B-VIII.0.

CONCLUSIONS

A sensitivity study was performed to investigate the variation of bias and bias uncertainties with respect to cooling time for PWR assemblies with increased enrichment and burnup. Although prior evidence suggested that such variation was minimal and overshadowed by the uncertainties in the bias, cooling time was one parameter with an established behavior and was thus chosen for further investigation.

No variation in the EALF-trended biases was observed with ENDF/B-VII.1, neither as a function of Pu content nor system EALF. Although this finding may seem to conflict with findings in NUREG/CR-7109, it should be noted again that trend values presented herein are the result of EALF trends, not directly trending on Pu content. Additionally, the bias behavior in ENDF/B-VIII.0 demonstrates the minor variability with data library.

A small trend was observed for the bias uncertainty, varying ~20 pcm over 40 years. This trend is due to the spectral hardening of the fuel with increased cooling, with a coinciding increase in Pu content. The hardened spectrum results in more distance between the trend at an evaluated value and the centroid of applicable experiment EALFs.

NUREG/CR-7109 demonstrated the stability of bias and bias uncertainties with respect to most burnup credit parameters. This study demonstrated through EALF trending the effective stability of the bias and bias uncertainty with respect to cooling time and data library.

REFERENCES

1. M.D. DEHART, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," ORNL/TM-12973, Oak Ridge National Laboratory, Oak Ridge, TN, May 1996, https://doi.org/10.2172/814237.
2. J. M. SCAGLIONE, et al., "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Analyses—Criticality (keff) Predictions," NUREG/CR-7109 (ORNL/TM-2011/514), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, April 2012.
3. A. ALPAN, et al., "Fuel and Irradiation Parametric Study for Extended Enrichment and High Burnup Light Water Reactor Spent Nuclear Fuel in Dry Storage Casks and Transportation Packages," NUREG, prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, in review.
4. W. METWALLY, et al., "Validation Studies for High Burnup and Extended Enrichment Fuels in Burnup Credit Criticality Safety Analyses," NUREG, prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, in review.
5. J. C. WAGNER, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit," NUREG/CR-6747 (ORNL/TM-2000/306), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, October 2001.
6. W. WIESELQUIST, R. LEFEBVRE, "SCALE 6.3.0 User Manual," ORNL/TM-SCALE-6.3.1. Oak Ridge National Laboratory, Oak Ridge, TN (2023).
7. M. B. CHADWICK, et al., "ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data," Nuclear Data Sheets, Vol. 112, No. 12, pp. 2887-2996, December 2011,
8. D. A. BROWN, et al., "ENDF/B-VIII.0: The 8th Major Release of the Nuclear Reaction Data Library with CIELO-project Cross Sections, New Standards and Thermal Scattering Data," Nuclear Data Sheets, Vol. 148,pp. 1-142, February 2018,
9. J. C. DEAN, R. W. TAYLOE Jr., and D. Morey, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, prepared for the US Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, January 2001.