ADDRESSING FISSION PRODUCT VALIDATION IN MCNP BURNUP CREDIT CRITICALITY CALCULATIONS

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ABSTRACT

The US Nuclear Regulatory Commission (NRC) Division of Spent Fuel Storage and Transportation issued Interim Staff Guidance (ISG) 8, Revision 3 in September 2012. This ISG provides guidance for NRC staff members' review of burnup credit (BUC) analyses supporting transport and dry storage of pressurized water reactor spent nuclear fuel (SNF) in casks. The ISG includes guidance for addressing validation of criticality (k_{eff}) calculations crediting the presence of a limited set of fission products and minor actinides (FP&MAs).

Based on previous work documented in NRC Regulatory Guide (NUREG) Contractor Report (CR)-7109, the ISG recommends that NRC staff members accept the use of either 1.5 or 3% of the FP&MA worth—in addition to bias and bias uncertainty resulting from validation of k_{eff} calculations for the major actinides in SNF—to conservatively account for the bias and bias uncertainty associated with the specified unvalidated FP&MAs. The ISG recommends (1) use of 1.5% of the FP&MA worth if a modern version of SCALE and its nuclear data are used and (2) 3% of the FP&MA worth for well qualified, industry standard code systems other than SCALE with the Evaluated Nuclear Data Files, Part B (ENDF/B),-V, ENDF/B-VI, or ENDF/B-VII cross sections libraries.

The work presented in this paper provides a basis for extending the use of the 1.5% of the FP&MA worth bias to BUC criticality calculations performed using the Monte Carlo N-Particle (MCNP) code. The extended use of the 1.5% FP&MA worth bias is shown to be acceptable by comparison of FP&MA worths calculated using SCALE and MCNP with ENDF/B-V, -VI, and -VII–based nuclear data. The comparison supports use of the 1.5% FP&MA worth bias when the MCNP code is used for criticality calculations, provided that the cask design is similar to the hypothetical generic BUC-32 cask model and that the credited FP&MA worth is no more than 0.1 Δk_{eff} (ISG-8, Rev. 3, Recommendation 4).

KEYWORDS
burnup credit, BUC, validation

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1. INTRODUCTION

The concept of taking credit for the reduction in reactivity resulting from the net consumption of fissile nuclides and creation of neutron absorbing actinides and fission products during reactor operation is commonly referred to as *burnup credit* (BUC). The analysis supporting BUC requires two different types of computer calculations. First, the analyst uses a computer code to generate spent nuclear fuel (SNF) compositions based on conditions experienced by the fuel assemblies in the reactor. The SNF compositions are then typically used in three-dimensional Monte Carlo method neutron transport calculations to determine the neutron multiplication factor (k_{eff}) values for the system loaded with SNF. The combination of computer codes, nuclear data, computational input options used, and modeling approximations used is referred to as the *computational method*.

Computational method validation is used to quantify the relationship between calculated values and reality and is performed by comparing results that are calculated using the computational method to the measured or expected results. For example, the computational method is used to model a set of laboratory critical experiments yielding calculated estimates of k_{eff}. Appropriate statistical techniques are applied to the calculated and expected k_{eff} values to determine the bias and bias uncertainty associated with the computational method. American National Standards Institute (ANSI) /American Nuclear Society (ANS)-8.1-2014 [1] and ANSI/ANS-8.24-2007 [2] address requirements and recommendations for validating criticality safety calculations. ANSI/ANS-8.27-2008 [3] provides additional validation guidance specific to BUC analyses.

Use of the same computational method in both the safety analysis and the validation study enables the analyst to quantify the bias and bias uncertainty associated with the computational method. However, the accuracy of the calculated bias and bias uncertainty is also dependent on the degree of similarity between the safety analysis models and the critical experiments. If materials are present in the safety analysis models but not the critical experiments, or vice versa, then any bias associated with such materials may not be captured correctly by the validation study. Bias errors also can be introduced when a material is present in both the safety analysis models but not the critical experiments, but this has a different effect on k_{eff}. The same nuclear data errors then affect the k_{eff} of the safety analysis models and critical experiments differently, resulting in different k_{eff} biases. Such differences may result from factors such as shifts in the neutron energy spectrum to higher or lower energies, or they may be due to spatial variation of the neutron flux.

The BUC analyses for storage and transportation casks frequently seek to take credit for some fission products (FP) and actinides for which little or no appropriate critical experiment data are freely available for use in validation studies. At the request of the US Nuclear Regulatory Commission (NRC), Oak Ridge National Laboratory (ORNL) staff members prepared report NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions* [4], which recommended methods to address poor validation of fission products and minor actinides (FP&MA) in k_{eff} calculations. Based in part on the uncertainty analysis work presented in NUREG/CR-7109, the NRC Division of Spent Fuel Storage and Transportation provided the following guidance in Recommendation 4 of Interim Staff Guidance (ISG) 8, Rev. 3 [5] (some supplementary explanation is provided within brackets in the following text):

> Fission product and minor actinide credit
>
> *The applicant may credit the minor actinide and fission product nuclides listed in Table 2* [ISG-8, Rev. 3; Table 2 lists the FP&MA shown in Table I in this paper], *provided the bias and bias uncertainty associated with the major actinides is determined as described above* [ISG-8, Rev. 3 provides guidance for validation of major actinides]. *One point five percent (1.5%) of the worth*

of the minor actinides and fission products conservatively covers the bias due to these isotopes. Due to the conservatism in this value, no additional uncertainty in the bias needs to be applied. This estimate is appropriate provided the applicant:

- *uses the SCALE code system with the ENDF/B–V, ENDF/B–VI, or ENDF/B–VII cross section libraries,*
- *can justify that its design is similar to the hypothetical GBC-32* [generic burnup credit cask–32, Ref. 8] *system design used as the basis for the NUREG/CR-7109 criticality validation, and*
- *demonstrates that the credited minor actinide and fission product worth is no greater than 0.1 in* k_{eff}.

For well qualified, industry standard code systems other than SCALE with the ENDF/B-V, ENDF/B–VI, or ENDF/B–VII cross section libraries, a conservative estimate for the bias associated with minor actinide and fission product nuclides of 3.0% of their worth may be used. Use of a minor actinide and fission product bias less than 3.0% should be accompanied by additional justification that the lower value is an appropriate estimate of the bias associated with that code system.

The FP&MA nuclides covered by the ISG-8, Rev. 3 guidance are listed in Table 2 of that document and are provided below in Table I.

Table I. Fission products and minor actinides

Fission products			
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{143}Nd	^{145}Nd
^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd
Minor actinides			
^{236}U	^{237}Np	^{243}Am	

The purpose of the work presented in this paper was to investigate and establish a basis for extending use of the 1.5% of the FP&MA worth bias to criticality calculations performed using the Monte Carlo N‑Particle (MCNP) computer code [6,7] if appropriate. NRC publication of the information in this paper as a NRC Regulatory Guide (NUREG) / contractor report (CR) is expected before the end of calendar year 2015.

2. FISSION PRODUCT AND MINOR ACTINIDE WORTHS – METHODS AND RESULTS

Development of a basis for extending the applicability of the 1.5% FP&MA worth bias to MCNP calculations involved calculating FP&MA worths for SNF in a cask geometry using MCNP and KENO and cross sections based on various ENDF releases. The FP&MA worths are then compared to determine whether the MCNP FP&MA worths compare well or conservatively with the worths calculated using KENO. Since full computational method validation of the remainder of the system (i.e., other than the FP&MA) is still required, it is acceptable to focus solely on comparison of the FP&MA worths.

The following subsections describe the analytical approach, SNF compositions used, criticality models used, the FP&MA worth results, and comparisons of the results.

2.1. Analytical Methods

The analytical method described in this section is used to demonstrate that MCNP used with its ENDF/B-V, -VI, -VII or -VII.1–based data sets produces FP&MA worth results similar to those generated using the SCALE 6.1 code system with the ENDF/B-VII based nuclear data library.

The method used is as follows:

1. SCALE Criticality Safety Analysis Sequence–5 (CSAS-5) [8] and MCNP criticality calculations are performed for the GBC-32 cask [9] loaded with burned pressurized water reactor (PWR) fuel at 20 and 40 gigawatt days per metric ton of uranium (GWd/MTU) burnups and with post-irradiation cooling times of 5 and 40 years. Criticality calculations are performed with and without the FP&MA nuclides present. Descriptions of the cask and fuel assembly geometries and materials are provided in Section 2.2.

   MCNP calculations are performed using MCNP5 (ver. 1.60) and MCNP6.

2. Results from criticality calculations performed with and without the FP&MAs are used to calculate the FP&MA worth for each burnup and cooling time combination. The FP&MA worths are calculated by subtracting the k_{eff} calculated with the FP&MAs present from the k_{eff} calculated with FP&MA removed.

3. The calculated FP&MA worth results are then compared to FP&MA reference results which were calculated consistently with the computational method used to generate NUREG/CR-7109—using SCALE 6.1 and its ENDF/B-VII 238 neutron energy group library. Comparisons are made at 20 and 40 GWd/MTU burnups following 5- and 40-year post-irradiation cooling times to demonstrate that the comparison is valid over a range of safety analysis model parameters.

Underprediction of FP&MA worth leads to the calculation of a smaller FP&MA bias term, but the reduced FP&MA worth has a significantly larger and more conservative impact on the maximum k_{eff} through the calculated k_{eff} value than does the reduction in the uncertainty associated with the 1.5% of FP&MA worth bias term. Consequently, variation of FP&MA worth below the FP&MA reference values is considered conservative. Also, it is considered reasonable to accept the use of FP&MA worths that do not exceed the reference results by more than 1.5%.

Subject to additional considerations prescribed by ISG-8, Rev. 3 and described earlier in this section, it is recommended that the 1.5% of the FP&MA worth uncertainty be used as a bias term in MCNP-based k_{eff} analyses to cover the poor validation of FP&MA nuclides in MCNP k_{eff} calculations.

2.2. Calculation Models

NUREG/CR-7109 provides a complete description of a GBC-32 model loaded with Westinghouse 17 × 17 optimized fuel assemblies. The GBC-32 cask model is a representative PWR fuel storage cask that has been used in many sensitivity studies at ORNL. It was originally defined in NUREG/CR-6747 [9].

Burned fuel compositions are generated using ORIGEN through the STARBUCS sequence, along with the same in-reactor depletion-specific ORIGEN libraries that were used in the work reported in

NUREG/CR-7109. Compositions are generated for fuel that had an initial enrichment of 3.1 wt % ^{235}U, that is burned to 20 GWd/MTU, and that included 5- and 40-year post-irradiation cooling times. Compositions are also generated for 4.7 wt % ^{235}U fuel burned to 40 GWd/MTU with 5- and 40-year cooling times.

To simplify the calculations, single axial zone models are used for the determination of FP&MA worth. This is done to show that given the same models and compositions, the MCNP FP&MA worths will be similar to the FP&MA worths calculated using SCALE.

Figures 1 and 2 show the GBC-32 model used for this work, and Table II presents the burned fuel compositions used.

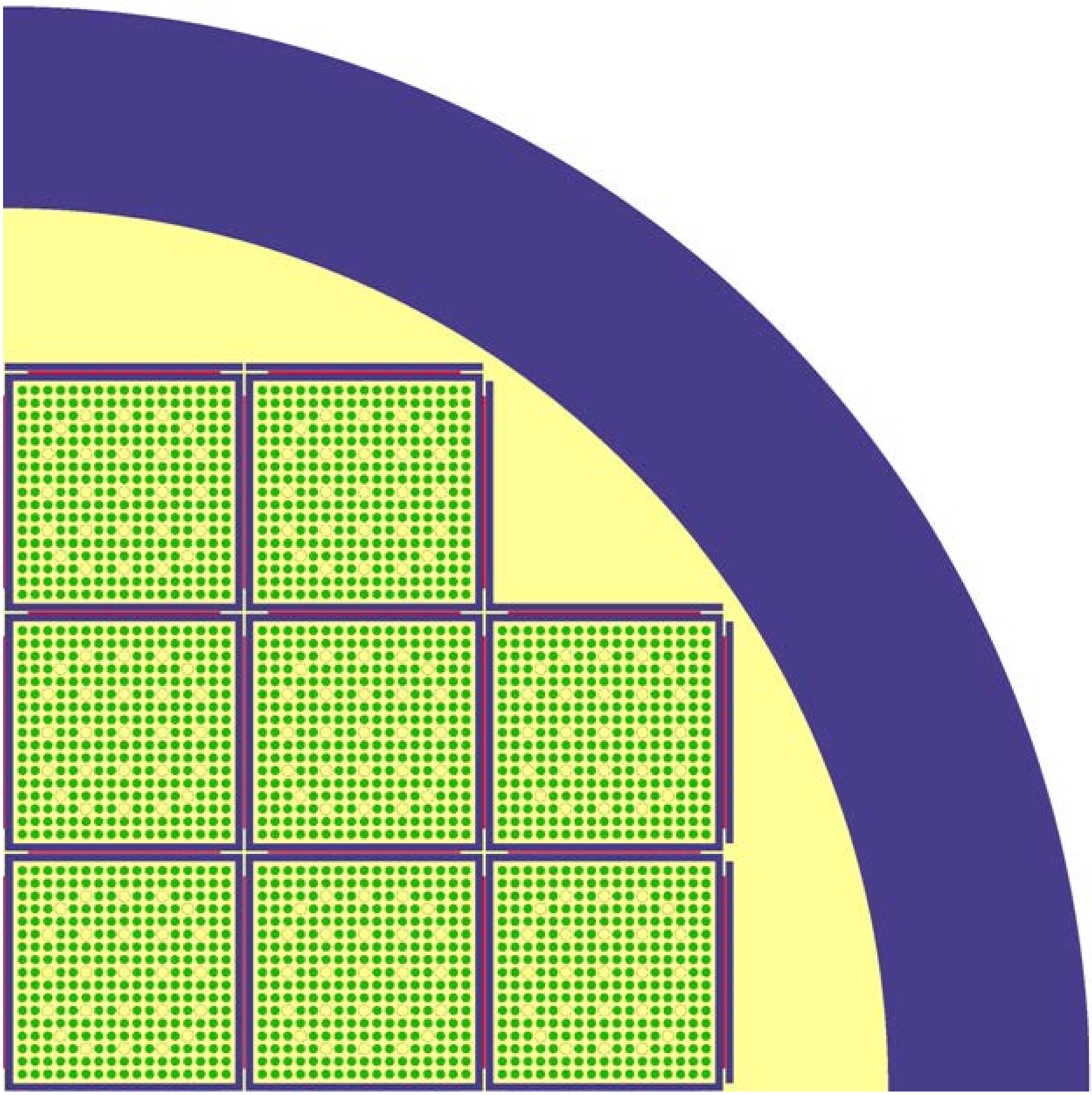

Figure 1. Radial Cross Section View of One Quarter of the GBC-32 Cask Model.

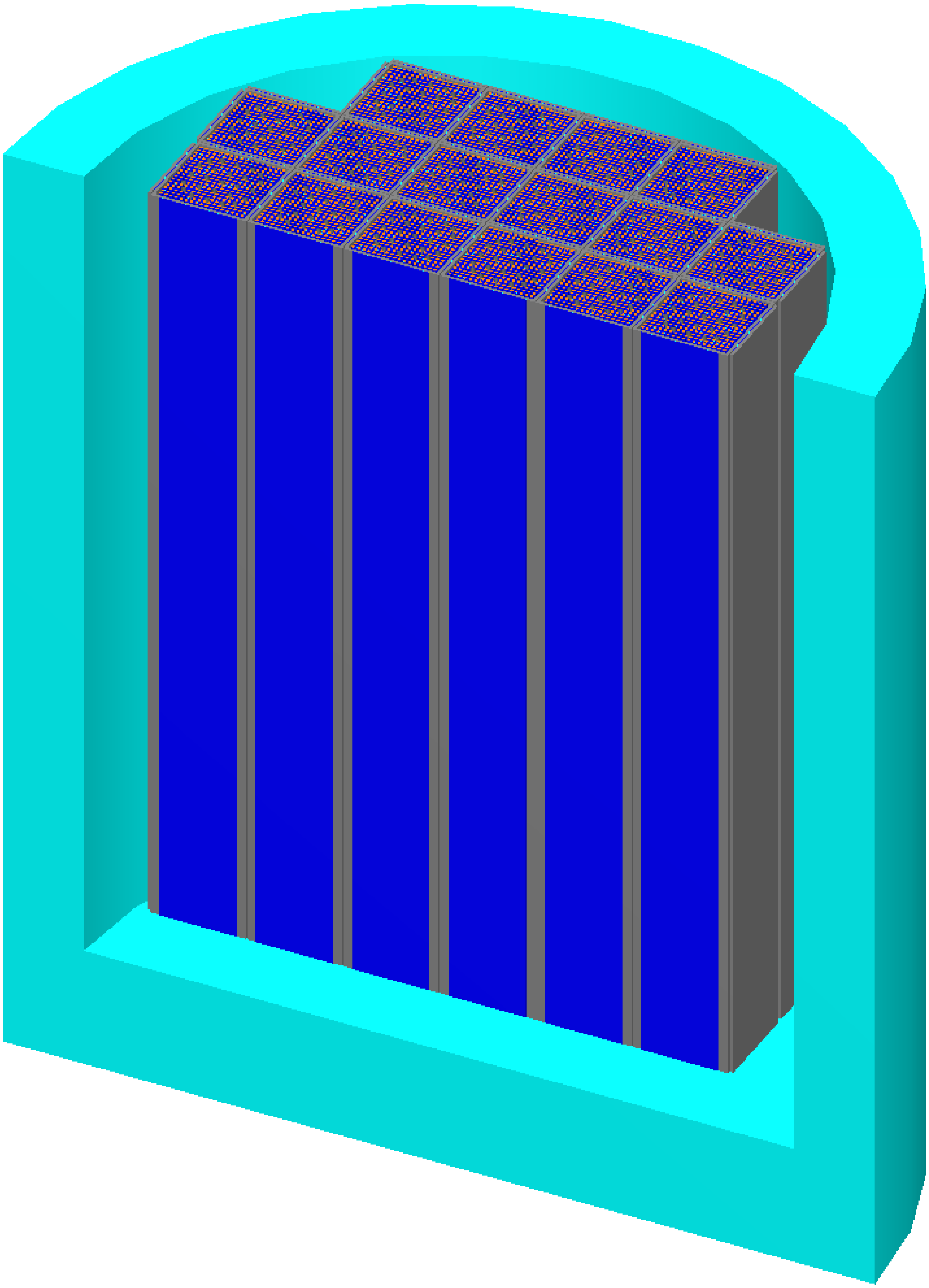

Figure 2. Isometric View of GBC-32 Cask with Top and Front Halves Removed.

Table II. Burned fuel compositions used

	Initial enrichment and final burnup	3.1 wt % and 20 GWd/MTU		4.7 wt % and 40 GWd/MTU	
	Cooling times	5 years	40 years	5 years	40 years
	Nuclides	Atom densities (atoms/barn-cm)			
Major actinides	^{234}U	7.3901E-08	3.2108E-07	2.6245E-07	1.3559E-06
	^{235}U	3.6046E-04	3.6060E-04	3.6741E-04	3.6758E-04
	^{238}U	2.2345E-02	2.2345E-02	2.1683E-02	2.1683E-02
	^{238}Pu	1.0223E-06	7.7547E-07	4.5238E-06	3.4316E-06
	^{239}Pu	1.3719E-04	1.3705E-04	1.7413E-04	1.7396E-04
	^{240}Pu	3.9569E-05	3.9477E-05	6.3031E-05	6.3301E-05
	^{241}Pu	1.6819E-05	3.0797E-06	3.2475E-05	5.9464E-06
	^{242}Pu	3.8879E-06	3.8877E-06	1.1666E-05	1.1665E-05
	^{241}Am	4.8440E-06	1.7840E-05	9.7110E-06	3.4784E-05
Minor actinides	^{236}U	7.0106E-05	7.0252E-05	1.3508E-04	1.3532E-04
	^{237}Np	5.5048E-06	6.2485E-06	1.3907E-05	1.5362E-05
	^{243}Am	5.1541E-07	5.1371E-07	2.5940E-06	2.5854E-06
Fission products	^{95}Mo	2.8435E-05	2.8435E-05	5.4717E-05	5.4717E-05
	^{99}Tc	2.9026E-05	2.9023E-05	5.5148E-05	5.5142E-05
	^{101}Ru	2.6347E-05	2.6347E-05	5.1618E-05	5.1618E-05
	^{103}Rh	1.8264E-05	1.8264E-05	3.1900E-05	3.1900E-05
	^{109}Ag	2.1375E-06	2.1375E-06	4.3961E-06	4.3961E-06
	^{133}Cs	3.0552E-05	3.0552E-05	5.7163E-05	5.7163E-05
	^{143}Nd	2.2819E-05	2.2819E-05	4.1309E-05	4.1309E-05
	^{145}Nd	1.6695E-05	1.6695E-05	3.1395E-05	3.1395E-05
	^{147}Sm	5.6742E-06	7.4723E-06	9.3467E-06	1.1950E-05
	^{149}Sm	1.7982E-07	1.7982E-07	2.2875E-07	2.2875E-07
	^{150}Sm	6.4540E-06	6.4540E-06	1.3517E-05	1.3517E-05
	^{151}Sm	4.1578E-07	3.1754E-07	6.4777E-07	4.9471E-07
	^{152}Sm	2.6638E-06	2.6639E-06	4.2868E-06	4.2871E-06
	^{151}Eu	1.6613E-08	1.1485E-07	2.5993E-08	1.7905E-07
	^{153}Eu	2.2518E-06	2.2518E-06	4.9812E-06	4.9812E-06
	^{155}Gd	7.0661E-08	1.3548E-07	1.7686E-07	3.3841E-07
	^{16}O	4.6943E-02	4.6943E-02	4.6949E-02	4.6949E-02
	Totals	7.0120E-02	7.0122E-02	6.9808E-02	6.9811E-02

2.3 Computational Results and Comparisons

The nuclear data uncertainty analysis results, which are presented in NUREG/CR-7109 and serve as the basis for ISG-8, Rev. 3, Recommendation 4, were generated using the SCALE 6.1 CSAS5 sequence and the ENDF/B-VII–based 238-neutron energy group library (v7-238). In that work, the CSAS5 sequence used the CENTRM module to perform problem-specific resolved resonance calculations, and KENO V.a Monte Carlo method neutron transport calculations were also performed.

The same computational method was used to generate the reference FP&MA worth results, which are presented in Table III. The models described in Section 2.2, including the burned fuel composition information provided in Table II, were used to calculate the reference FP&MA worths. Nominal calculations were performed with all of the nuclides listed in Table II present. Individual calculations were also performed with each of the FP&MAs removed, with all FP&MAs removed, with all minor actinides (MAs) removed, and with all fission products (FPs) removed. The group or individual nuclide worth values were then calculated as the change in k_{eff} caused by the presence of the nuclide or group of nuclides using the following equation:

$$\text{worth}_{\text{nuclide or group}} = k_{\text{nuclide or group removed}} - k_{\text{nominal}}$$

The results presented in the “All FP&MA,” “All MA,” and “All FP” rows of Table III were calculated with the group constituents simultaneously removed. The reference FP&MA worth results to be used for comparison with results from other computational methods are the worths listed in the “All FP&MA” row. The other worth values are provided for reference and to facilitate more detailed comparisons.

Table III. Reference FP&MA worth results using SCALE 6.1 and the ENDF/B-VII.0 238 group nuclear data library

Initial enrichment and final burnup	**3.1 wt % and 20 GWd/MTU**		**4.7 wt % and 40 GWd/MTU**	
Cooling time (years)	**5**	**40**	**5**	**40**
	Reactivity worth values (Δk)[a]			
All FP&MA (reference worth)	**0.06742**	**0.06921**	**0.10203**	**0.10442**
All MA	0.00602	0.00588	0.01154	0.01093
All FP	0.06058	0.06206	0.08820	0.09085
^{236}U	0.00373	0.00359	0.00579	0.00526
^{237}Np	0.00193	0.00210	0.00488	0.00459
^{243}Am	0.00022	0.00020	0.00128	0.00108
^{95}Mo	0.00123	0.00078	0.00207	0.00158
^{99}Tc	0.00278	0.00254	0.00439	0.00370
^{101}Ru	0.00088	0.00072	0.00165	0.00155
^{103}Rh	0.00681	0.00616	0.01023	0.00934
^{109}Ag	0.00105	0.00058	0.00180	0.00154
^{133}Cs	0.00324	0.00309	0.00551	0.00490
^{143}Nd	0.00895	0.00864	0.01348	0.01269
^{145}Nd	0.00187	0.00163	0.00332	0.00279
^{147}Sm	0.00139	0.00157	0.00223	0.00241
^{149}Sm	0.01570	0.01502	0.01646	0.01541
^{150}Sm	0.00095	0.00092	0.00165	0.00159
^{151}Sm	0.00588	0.00440	0.00760	0.00562
^{152}Sm	0.00233	0.00199	0.00323	0.00286
^{151}Eu	0.00027	0.00107	0.00043	0.00157
^{153}Eu	0.00150	0.00133	0.00309	0.00260
^{155}Gd	0.00359	0.00659	0.00738	0.01330

[a] Monte Carlo uncertainty (1 σ) is 0.00014 Δk and is not included in the FP&MA worths.

FP&MA reactivity worth calculations were performed using MCNP5 version 1.60 and MCNP6 with available continuous-energy nuclear data libraries including ENDF/B-V, -VI, -VII.0 and -VII.1. FP&MA worth results for these data sets and codes are provided in Table IV.

Table IV. Calculated FP&MA worths using various codes and data sets

Initial enrichment and final burnup		3.1 wt % and 20 GWd/MTU		4.7 wt % and 40 GWd/MTU	
Cooling time (years)		5	40	5	40
Code	Nuclear data[a]	FP&MA worths (Δ)[b] and % change from reference			
SCALE 6.1 (CENTRM) (reference worth)	**v7-238**	**0.06742**	**0.06921**	**0.10203**	**0.10442**
MCNP5 (1.60)	ENDF/B-VII.0	0.06748	0.06969	0.10213	0.10456
	(% change)[c]	+0.1	+0.7	+0.1	+0.1
MCNP6	ENDF/B-VII.0	0.06762	0.06961	0.10222	0.10450
	(% change)[c]	+0.3	+0.6	+0.2	+0.1
MCNP6	ENDF/B-VII.1	0.06663	0.0684	0.10099	0.10268
	(% change)[c]	-1.2	-1.2	-1.0	-1.7
MCNP5 (1.60)	ENDF/B-V	0.06688	0.06913	0.10154	0.10402
	(% change)[c]	-0.8	-0.1	-0.5	-0.4
MCNP6	ENDF/B-V	0.06705	0.06906	0.10136	0.10395
	(% change)[c]	-0.6	-0.2	-0.7	-0.5
SCALE 6.1 (CENTRM) (reference worth)[d]	**v6-238**	**0.04518**	**0.04923**	**0.06939**	**0.07535**
MCNP5 (1.60)[d]	ENDF/B-VI	0.04529	0.04918	0.0691	0.07487
	(% change)[c]	+0.2	-0.1	-0.4	-0.6
MCNP6[d]	ENDF/B-VI	0.04525	0.04908	0.0692	0.07511
	(% change)[c]	+0.2	-0.3	-0.3	-0.3

[a] SCALE nuclear data libraries are described in Section M4 of Ref.[8]. MCNP nuclear data libraries are described in Ref. [6] and [7].
[b] The Monte Carlo one-standard deviation uncertainty associated with all reported FP&MA worths is no greater than 0.00015 Δk. This uncertainty is not included in the FP&MA worths.
[c] Percent change from reference worth.
[d] Compositions excluded ^{95}Mo, ^{101}Ru, ^{143}Nd, ^{145}Nd, ^{150}Sm, ^{151}Sm, and ^{152}Sm, for which ENDF/B-VI data were not available in MCNP5 (v1.60) and MCNP6.

The 1.5% of the FP&MA worth bias term specified in Recommendation 4 of ISG-8, Rev. 3 was based in general on the work reported in NUREG/CR-7109 (Ref.4) and, more specifically, on the work reported in Section 7.4.3 of Ref. [4]. In that work, the uncertainty information associated with the nuclear data was combined with the nuclide-, reaction-, and energy-dependent sensitivity of k_{eff} to nuclear data variation for representative spent fuel pool and BUC cask models to generate the uncertainty in k_{eff} due to the uncertainty in nuclear data. The uncertainties in k_{eff} due to FP&MA were then compared to their worths, yielding an estimate that 1.5% of the FP&MA worth would yield a bounding estimate for the biases associated with the FP&MAs in the GBC-32 model. Information concerning the analysis technique and supporting the determination of the 1.5% value is presented in NUREG/CR-7109.

Without exception, the results obtained using MCNP vary from the reference values by less than 1.7% of the FP&MA reference worth value. This is expected because the variation between the results is solely due to nuclear data variation or differences in the implementation of the Monte Carlo calculation in SCALE CSAS versus MCNP.

Underprediction of FP&MA worth leads to the calculation of a smaller FP&MA bias term. However, the reduced FP&MA worth has a significantly larger and more conservative impact on the maximum k_{eff} through the calculated k_{eff} value than does the reduction in the 1.5% of FP&MA worth bias term. Consequently, variation from reference values is tolerable if the calculated FP&MA worths do not exceed the reference results by more than 1.5% of the FP&MA worths.

These results support use of the 1.5% of FP&MA worth bias term described in ISG-8, Rev. 3, Recommendation 4, when MCNP and the nuclear data sets described in Table IV are used in safety analyses. It may be possible to use the same method and comparisons to justify application of the 1.5% or 3.0% of FP&MA worth biases to results generated using other codes and/or nuclear data.

Some reference worths provided in Table IV appear to exceed the restriction provided in Recommendation 4 of ISG-8, Rev. 3, that the credited FP&MA worth not exceed 0.1 Δk_{eff}. Some of the FP&MA worths reported in this work are higher than those reported in NUREG/CR-7109 because of the use of the single-axial zone model, which amplifies the importance of the higher levels of FP&MAs in the center of the fuel. Further, when the number of significant figures is considered, the 0.10456 Δk_{eff} value does not exceed 0.1. Thus, if the credited FP&MA worth is slightly higher than 0.1 Δk_{eff}, the credited worth should be considered to be no greater than 0.1 Δk_{eff}, meeting the ISG-8, Rev. 3 criterion.

Note that all MCNP calculations were performed with the MCNP5, v1.60, and MCNP6 codes and nuclear data distributed by the MCNP developers. Since the neutron absorption reaction in FP&MA is the only significant interaction of neutrons with FP&MA nuclides, and the simulation of neutron capture is straightforward, the version of MCNP used does not affect the FP&MA worth results. Review of the data in Table IV reveals that FP&MA worths calculated with MCNP5 and MCNP6 using the same nuclear data set are statistically the same (i.e., they vary by less than one or two standard deviations). Consequently, the conclusion concerning the application of the SCALE FP&MA uncertainty information to MCNP calculation k_{eff} values is not MCNP version specific. However, the conclusions may not be applicable to special MCNP versions and/or data that were not generated by the MCNP developers.

3. CONCLUSIONS

The criticality safety of SNF in transportation or storage systems relies on the accurate calculation of the k_{eff} values. Validation studies are used to establish the relationship between the actual and calculated k_{eff} values. Unfortunately, there are not sufficient critical experiment data available to support the use of the conventional validation approach for FP&MA in BUC criticality safety evaluations for PWR SNF casks. Work documented in NUREG/CR-7109 (Ref.4) supports the use of nuclear data uncertainty to provide a bounding estimate of the potential bias associated with taking credit for FP&MA in BUC criticality analyses. Based on that work, the NRC Division of Spent Fuel Storage and Transportation Interim Staff Guidance 8, Rev. 3, Recommendation 4, provides guidance for adoption of a bias term equal to 1.5% or 3.0% of the FP&MA worth, depending on the code and nuclear data used.

The work documented in this paper provides justification for use of 1.5% of the FP&MA worth as a bias in BUC criticality safety evaluations of intact PWR fuel assemblies using MCNP with the ENDF/B-V, -VI , -VII.0 or -VII.1 nuclear data distributed with those code systems by the MCNP development team. For other code systems or nuclear data sets, it may be possible to use the same method to confirm that their computational method yields FP&MA worths similar to those calculated using the SCALE 6.1 CSAS5 sequence and the ENDF/B–VII 238 neutron energy group nuclear data library.

Results generated using MCNP with multiple sets of nuclear data are presented in Section 2. All nuclear data sets examined yielded FP&MA worths that were within 1.7% of the reference FP&MA worths. The MCNP results were no greater than 0.7% larger than the reference FP&MA worths for any of the burnup, decay time, or nuclear data libraries considered. Consequently, use of the 1.5% of FP&MA worth bias to account for poor validation of FP&MAs in criticality calculations performed using MCNP5 or MCNP6 with ENDF/B-V, -VI, -VII.0 or -VII.1 data is recommended.

NRC publication of the information and recommendations in this paper as a NUREG/CR report is expected before the end of calendar year 2015.

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REFERENCES

1. ANSI/ANS-8.1-2014, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,” American Nuclear Society (2014).
2. ANSI/ANS-8.24-2007(R2012), “Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations,” American Nuclear Society (2007).
3. ANSI/ANS-8.27-2008, “Burnup Credit for LWR Fuel,” American Nuclear Society (2008).
4. 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 US Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee (2012).
5. Division of Spent Fuel Storage and Transportation, Interim Staff Guidance-8, Rev. 3, *Burnup Credit in the Criticality Analyses of PWR Spent Fuel in Transportation and Storage Casks*, US Nuclear Regulatory Commission, (2012).
6. *MCNP—A General Monte Carlo N-Particle Transport Code, Version 5*, LA-UR-03-1987, Los Alamos National Laboratory, Los Alamos, NM, February 1, 2008. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-740.
7. *Initial MCNP6 Release Overview – MCNP6 Version 1.0, Rev. 1*, LA-UR-13-22934, Los Alamos National Laboratory, Los Alamos, NM, June 27, 2013. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-810.
8. *Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.
9. 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 US Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee, (2001).