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Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks

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Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks

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ABSTRACT

The U.S. Nuclear Regulatory Commission Division of Spent Fuel Management issued Interim Staff Guidance (ISG) 8, Revision 3 in September 2012. This ISG provides guidance for NRC Staff review of burnup credit analyses supporting transport and dry storage of pressurized water reactor spent nuclear fuel (SNF) in casks. ISG-8, Rev. 3 includes, among other things, guidance for addressing validation of criticality (k_{eff}) calculations crediting the presence of a limited set of fission products and minor actinides (FP&MA). Based on previous work documented in NUREG/CR-7109, ISG-8, Rev. 3 includes a recommendation to accept use of 1.5 or 3% of the FP&MA worth, depending on the criticality code and cross-section data used, to conservatively account for the bias and bias uncertainty associated with the selected insufficiently validated FP&MAs. This bias is applied in addition to the bias and bias uncertainty resulting from validation of k_{eff} calculations for the major actinides in SNF. The work described in this report involves comparison of FP&MA worths calculated using SCALE and Monte Carlo N-Particle (MCNP) with Evaluated Nuclear Data Files, Part B–V, –VI, and –VII based nuclear data. The comparison supports use of the 1.5% FP&MA worth bias when either SCALE or MCNP codes are used for criticality calculations, provided the other conditions of ISG-8, Rev. 3, Recommendation 4 are met. The additional conditions include that the cask design is similar to the hypothetical generic burnup credit-32 cask model and that the credited FP&MA worth is no more than 0.1 Δk_{eff}.

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ACRONYMS

BUC	burnup credit
CENTRM	continuous energy transport module
CSAS	Criticality Safety Analysis Sequence
ENDF/B	Evaluated Nuclear Data Files, Part B
FP	fission products
FP&MA	fission products and minor actinides
GBC	generic burnup credit cask; as in GBC-32, a generic burnup credit cask model with a 32 PWR fuel assembly capacity
GWd/MTU	unit of nuclear fuel burnup; gigawatt-days per initial metric ton of uranium
ISG	interim staff guidance
k_{eff}	effective neutron multiplication factor
MA	minor actinides
MCNP	Monte Carlo N-Particle
NRC	U.S. Nuclear Regulatory Commission
NUREG/CR	Publications prepared by NRC Contractors
OFA	optimized fuel assembly; as in Westinghouse 17 × 17 OFA
ORNL	Oak Ridge National Laboratory
PWR	pressurized water reactor
SNF	spent nuclear fuel

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 spent fuel compositions are then typically used in three-dimensional Monte Carlo method neutron transport calculations to determine the k_{eff} values for the system loaded with SNF. The combination of the 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, calculated using the computational method, to measured or expected results. For example, the computational method would be used to model a set of laboratory critical experiments, yielding calculated estimates for the neutron multiplication factor (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.

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 either the safety analysis models or critical experiments and not in the other, any bias associated with these materials may not be captured correctly by the validation study. Bias errors also can be introduced when a material is present in both but 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 because of spatial variation of the neutron flux.

Burnup credit 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 available for use in validation studies. At the U.S. Nuclear Regulatory Commission's (NRC's) request, Oak Ridge National Laboratory (ORNL) staff prepared report NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions* (Ref. 1), which included recommendations for addressing 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 Management provided the following guidance in Recommendation 4 of Interim Staff Guidance (ISG) 8, Rev. 3 (Ref. 2) (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 [*of ISG-8, Rev. 3; Table 2 lists the FP&MA shown in Table 1.1 in this report*], 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 [*Evaluated Nuclear Data Files, Part 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*] 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 guidance are listed in Table 2 of ISG-8, Rev. 3, and below in Table 1.1.

Table 1.1. 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	

This report provides a complete description of a GBC-32 model loaded with pressurized water reactor (PWR) SNF. Single-axial zone burned fuel compositions are provided at fuel burnups of 20 and 40 GWd/MTU for post-irradiation cooling times of 5 and 40 years. Reference FP&MA worths are provided for the four burnup and cooling time combinations for several of the nuclear data libraries available in SCALE 6.1 (Ref. 3). Comparisons of SCALE 6.1 and Monte Carlo N‑Particle (MCNP) (Ref. 4) FP&MA worths are also provided.

Since the FP&MA worths calculated using MCNP are either lower than or statistically the same as the SCALE results (see Table 4.1), these comparisons demonstrate that it is appropriate to apply the 1.5% of the FP&MA worth bias to MCNP calculations using ENDF/B–V, –VI, –VII and –VII.1 data. This additional bias term supplements the MCNP k_{eff} calculation validation, thereby addressing the lack of k_{eff} validation for the FP&MAs.

1.1 CALCULATION METHOD

The calculation method described in this section demonstrates that MCNP used with its ENDF/B–V, –VI, –VII or –VII.1 based data sets produces FP&MA worth results that are similar to results generated using the SCALE 6.1 code system with ENDF/B–V, –VI and –VII based nuclear data libraries.

The calculation method used is the following:

1. SCALE Criticality Safety Analysis Sequence–5 (CSAS5) and MCNP criticality calculations are performed for the GBC-32 cask (Ref. 5) loaded with burned PWR fuel at 20 and 40 GWd/MTU burnups and with post-irradiation cooling times of 5 and 40 years. Complete descriptions of the cask and fuel assembly geometry and materials are provided in Section 2. Single axial zone burned fuel compositions are provided at burnup values of 20 and 40 GWd/MTU for 5- and 40-year post-irradiation cooling times. SCALE CSAS5 and MCNP cask models and code-specific (i.e., in code-specific input format) burned fuel composition inputs are provided in Appendixes A and B of this report.

   To simplify the calculations, single axial zone models are used for the determination of FP&MA worth. This is done because the goal is 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. 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.

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 FP&MA removed from the k_{eff} calculated with all FP&MAs present.

3. The calculated FP&MA worth results are then compared to FP&MA reference results presented in Table 3.1 in Section 3. The reference results were calculated, consistent with the computational method used to generate NUREG/CR-7109, using SCALE 6.1 and its ENDF/B–VII 238 energy group library. Detailed nuclide-by-nuclide k_{eff} worth results for other nuclear data libraries and for MCNP are provided in Appendix C.

Under prediction 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 reference FP&MA reference values is reasonable because the calculated FP&MA worths do not exceed the reference results FP&MA worths 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 in MCNP based k_{eff} analyses to cover the poor validation of FP&MA nuclides in MCNP k_{eff} calculations.

2 REFERENCE MODEL DESCRIPTION

The following subsections describe the reference model used to generate the FP&MA worths presented in Sections 3 and 4 and in Appendix C. The cask and SNF model used is the same as is described in NUREG/CR-6747 (Ref. 5). For the reader's convenience, relevant model information is provided in Sections 2.1 and 2.2. The burned PWR fuel compositions used in the reference model are described in Section 2.3. Input listings for the SCALE 6.1 CSAS5 and MCNP models of the GBC-32 cask are provided in Appendixes A and B, respectively.

2.1 GBC-32 CASK MODEL

As is discussed in NUREG/CR-6747, the GBC-32 cask model was developed as a computational benchmark that could be used for the estimation of the additional reactivity associated with FP&MA. The model was intended to capture the important features of transport and storage casks without use of non-essential details or proprietary information associated with actual cask designs. The cask holds 32 PWR assemblies in storage cells with 0.75-cm-thick steel walls as well as one Boral™ plate between each pair of assemblies and on the external faces of the basket.

The water-filled cask dimensions and materials are described in Tables Table 2.1 and Table 2.2. Figures Figure 2.1, Figure 2.2, and Figure 2.3 show features of the GBC-32 cask model. Tables Table 2.1 and Table 2.2, and Figures 2.1 and Figure 2.2 were reproduced from NUREG/CR-6747.

In addition to the information provided in Tables 2.1 and Table 2.2, the Boral™ panel present on the outside of each external fuel assembly storage cell is covered by a 0.75-cm-thick type 304 stainless steel plate. The width of each external vertical steel plate shown in Figure 2.1 matches the storage cell inside dimension of 22.0 cm. The width of each external horizontal steel plate matches the storage cell outside dimension of 23.5 cm.

Table 2.1. Physical dimensions for the GBC-32 cask

Parameter	Inches	Centimeters
Cell inside dimension	8.6614	22.0000
Cell outside dimension	9.2520	23.5000
Cell wall thickness	0.2953	0.7500
Boral™ panel thickness[a]	0.1010	0.2565
Boral™ center thickness	0.0810	0.2057
Boral™ Al plate thickness	0.0100	0.0254
Cell pitch	9.3530	23.7565
Boral panel width	7.5000	19.0500
Cell height[b]	144.0000	365.76
Boral™ panel height[b]	144.0000	365.76
Cask inside diameter	68.8976	175.0000
Cask outside diameter	84.6457	215.0000
Cask radial thickness	7.8740	20.0000
Base plate thickness	11.8110	30.0000
Cask lid thickness	11.8110	30.0000
Cask inside height	161.7165	410.7600
Active fuel height[b]	144.0000	365.76
Bottom assembly hardware thickness	5.9055	15.0000
Top assembly hardware thickness	11.8110	30.0000

[a] Boral™ is a clad composite of aluminum and boron carbide. A Boral™ panel or plate consists of three distinct layers. The outer layers are aluminum cladding, which form a sandwich with a central layer that consists of a uniform aggregate of fine boron carbide particles within an aluminum alloy matrix.
[b] The cell height, Boral™ panel height, and active fuel height are all equivalent, and their lower boundaries are coincident, 15 cm above the base plate.

Table 2.2. Material compositions for GBC-32 cask model

Isotope	Atom density (atoms/barn-cm)	Weight percent
Water (density = 0.9983 g/cm^3)		
Hydrogen (H)	0.06674	11.19
Oxygen (O)	0.03337	88.81
Total	0.10011	100.0
Stainless steel 304 (density = 7.92 g/cm^3) (Ref. 6)		
Chromium (Cr)	0.01743	19.0
Manganese (Mn)	0.00174	2.0
Iron (Fe)	0.05936	69.5
Nickel (Ni)	0.00772	9.5
Total	0.08625	100.0
Boral™ panel aluminum cladding (density = 2.699 g/cm^3)		
Aluminum (Al)	0.0602 (Ref. 7)	100.0
Total	0.0602	100.0
Boral™ panel central layer (0.0225 g ^{10}B/cm^2)[a]		
Boron-10 (^{10}B)	6.5794E-03	4.13
Boron-11 (^{11}B)	2.7260E-02	18.81
Carbon (C)	8.4547E-03	6.37
Aluminum (Al)	4.1795E-02	70.69
Total	8.4089E-02	100.0

[a] Note: 0.030 g ^{10}B/cm^2 is the loading from the manufacturer (AAR[8]) that corresponds to the modeled Boral panel thickness of 0.101 inches. However, current NRC guidance[9] recommends only 75% credit for fixed neutron absorbers, and thus 75% of 0.030, or 0.0225 g ^{10}B/cm^2 is used.

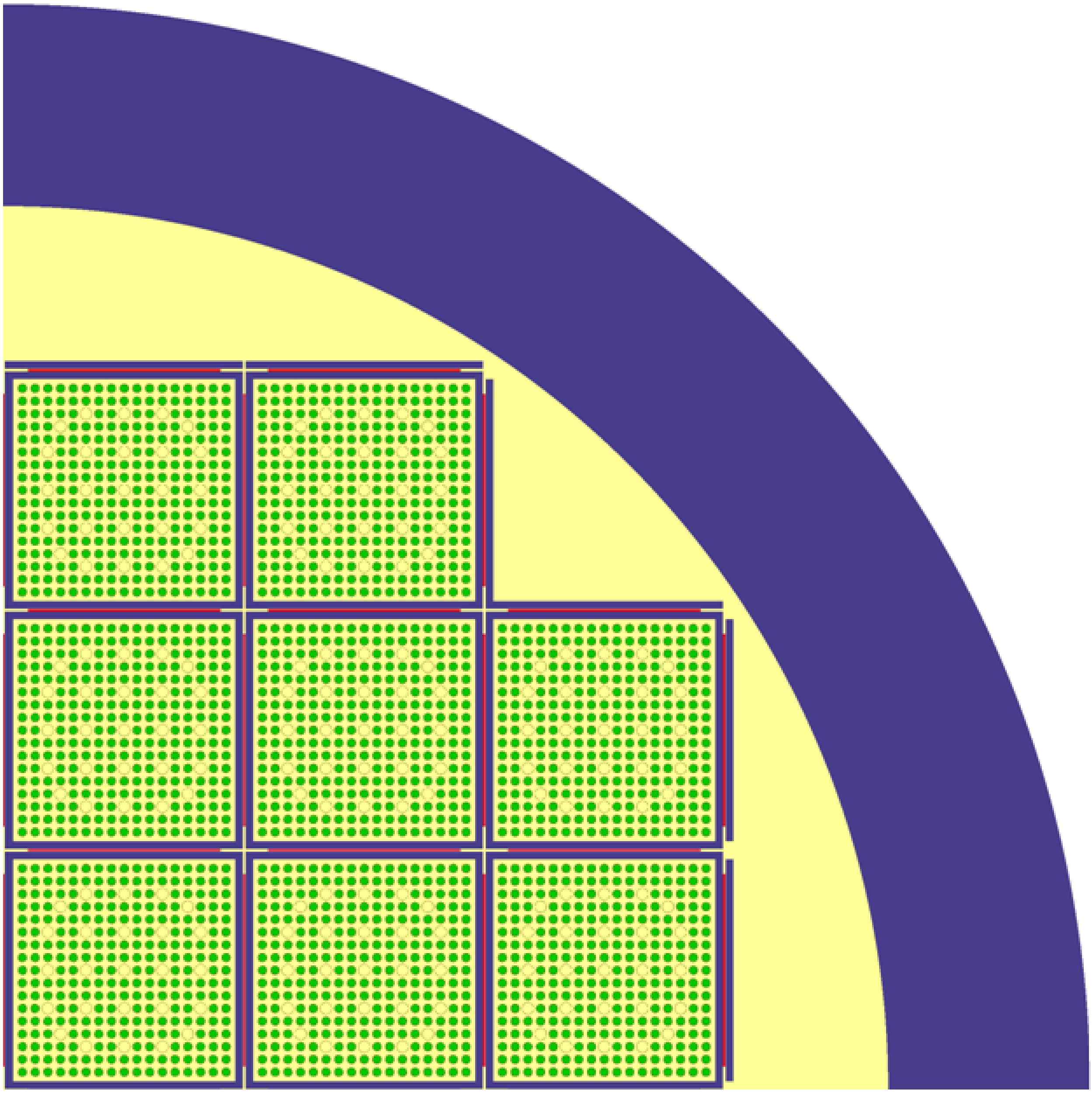

Figure 2.1. Radial cross section of one-quarter of the GBC-32 cask

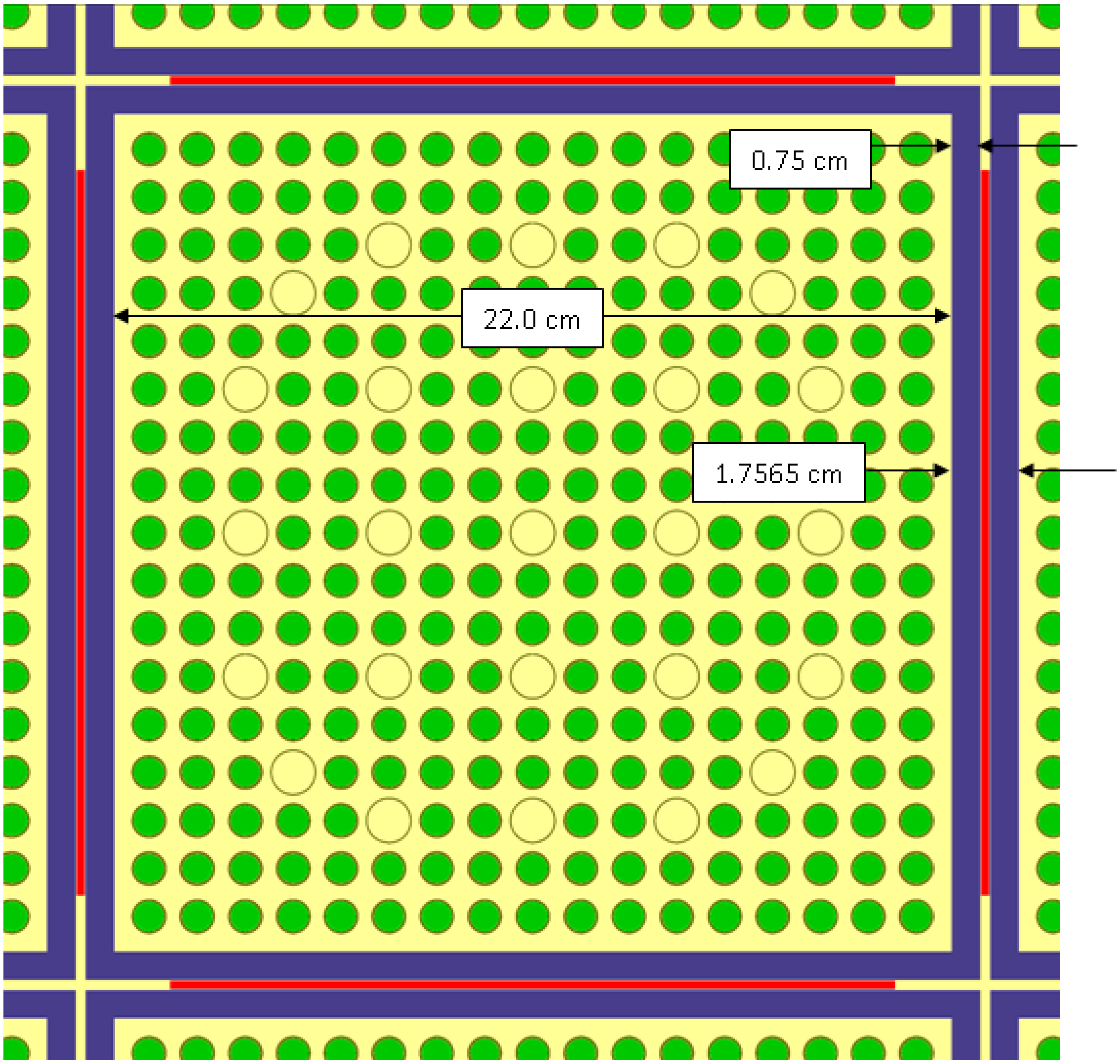

Figure 2.2. Cross-sectional view of an assembly cell in KENO V.a model for the GBC-32 cask

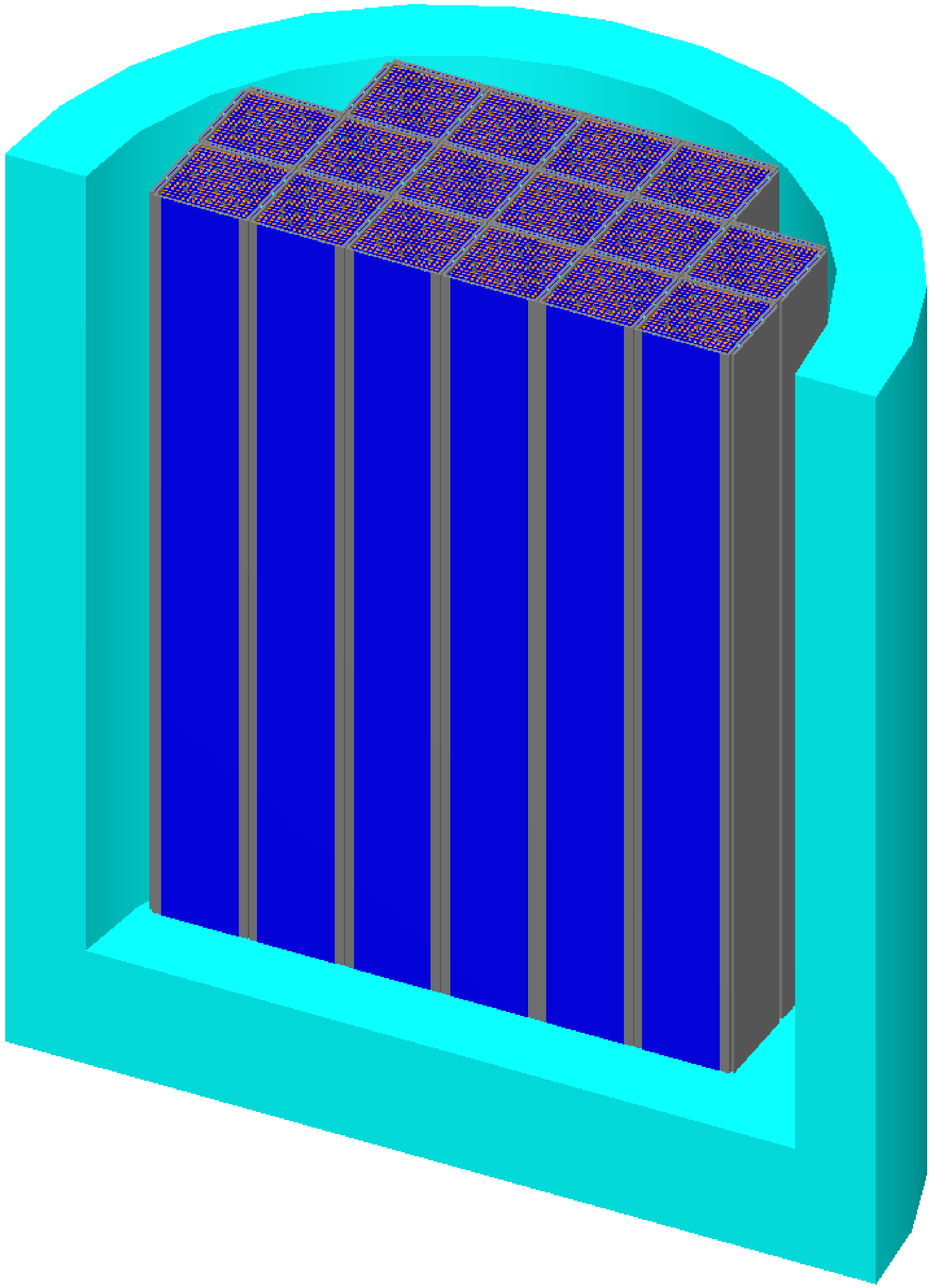

Figure 2.3. Isometric view of GBC-32 cask with top and front halves removed

2.2 PWR FUEL ASSEMBLY SPECIFICATION

The reference PWR fuel assembly design used in NUREG/CR-6747 and in NUREG/CR-7109 is representative of a Westinghouse 17 × 17 Optimized Fuel Assembly (OFA). To be consistent with prior work, the same assembly design is used in this report for calculation of the reference FP&MA worths. The simplified model used here and in NUREG/CR-6747 replaces the assembly grids and assembly structures above and below the active fuel length with water. Thus the simplified fuel assembly model is a 365.76-cm-long 17 × 17 array of fuel pins and guide tubes. Table 2.3, reproduced here from Table 3 of NUREG/CR-6747, provides the fuel assembly specifications.

The spent fuel pellet compositions will be addressed in the next section. The fuel rod clad, guide tubes, and instrument tube are all modeled as 100% zirconium, having an atom density of 0.0423 atoms per barn-cm (Ref. **Error! Bookmark not defined.**).

Table 2.3. PWR fuel assembly specifications

Parameter	Inches	Centimeters
Fuel outside diameter	0.3088	0.7844
Cladding inside diameter	0.3150	0.8001
Cladding outside diameter	0.3600	0.9144
Cladding radial thickness	0.0225	0.0572
Rod pitch	0.4960	1.2598
Guide tube/thimble inside diameter	0.4420	1.1227
Guide tube/thimble outside diameter	0.4740	1.2040
Thimble radial thickness	0.0160	0.0406
Instrument tube inside diameter	0.4420	1.1227
Instrument tube outside diameter	0.4740	1.2040
Instrument tube radial thickness	0.0160	0.0406
Active fuel length	144	365.76
Array size	17 × 17	
Number of fuel rods	264	
Number of guide tubes/thimbles	24	
Number of instrument tubes	1	

2.3 REFERENCE MODEL FUEL COMPOSITIONS

For the purposes of the reference FP&MA worth calculations, fuel compositions were calculated for two initial enrichment/final burnup pairs along a representative loading curve determined for the analysis documented in NUREG/CR-7109 (Ref. 1). This loading curve is the “Actinides & 16 FP” loading curve shown below in Figure 2.4, which was reproduced from Figure 4.4 of NUREG/CR-7109. The models used to generate the loading curves shown in the figure included an 18-axial zone burnup distribution and a 5-year post-irradiation cooling time. The background in this figure shows all PWR fuel inventory as of the end of 2002 according to the RW-859 (2002) data (Ref. 10).

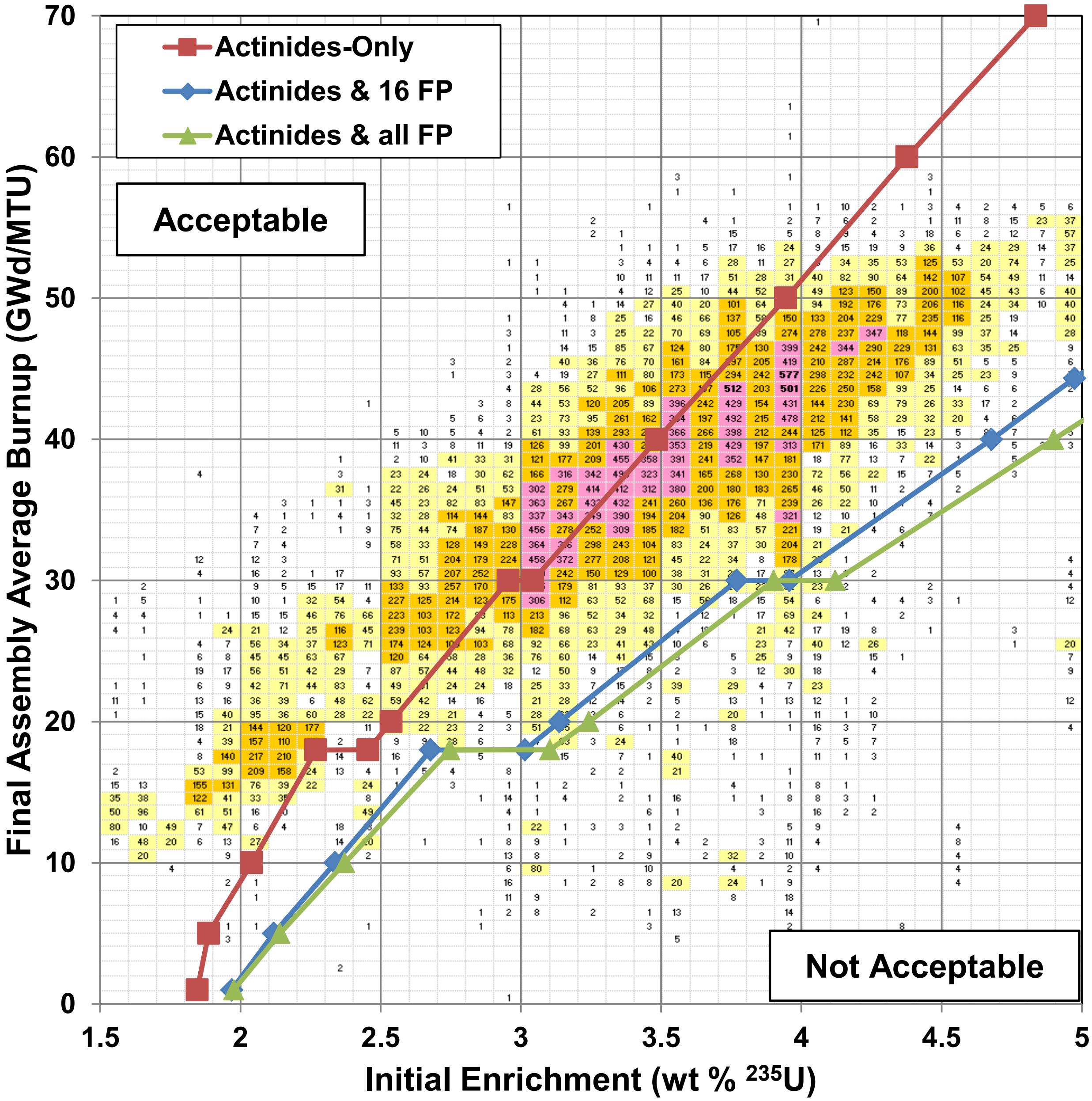

Figure 2.4. Burnup credit loading curves for the GBC-32 application models

In particular, the fuel compositions used in the reference FP&MA worth calculations are for fuel with initial enrichments of 3.14 and 4.68 wt % ^{235}U burned to 20 and 40 GWd/MTU, respectively. Fuel compositions were generated for each combination with both a 5-year and 40-year post-irradiation cooling time. Table 2.4 provides the burned fuel compositions used in the calculation of the reference FP&MA worth values. These irradiated fuel compositions were generated using the same computational methods described in NUREG/CR-7109 (Ref. 1).

Table 2.4. Burned fuel compositions for reference FP&MA worth calculations

	Initial enrichment and final burnup	3.1 wt % and 20 GWd/MTU		4.7 wt % and 40 GWd/MTU	
	Cooling time	5 years	40 years	5 years	40 years
	Nuclide	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
	Total	7.0120E-02	7.0122E-02	6.9808E-02	6.9811E-02

As has been shown in prior work (Ref. 11), accurate k_{eff} determination for PWR spent fuel systems requires that the axial burnup distribution be modeled. The calculations performed for this report used a single axially uniform fuel composition for each burnup and post-irradiation cooling time. This was done to simplify the calculations and is appropriate because the objective of the calculations is limited to showing that the other computational methods (e.g., MCNP with ENDF/B–V, –VI, –VII, and –VII.1 nuclear data) generate FP&MA worth results that are similar to the reference results. Calculations and comparisons were made at two burnup/enrichment points and for two post-irradiation cooling times to demonstrate that the comparisons made and conclusions reached are valid over the range of burned fuel compositions typically observed in burnup credit criticality analyses for transportation and dry storage operations.

3 REFERENCE FP&MA WORTHS

The nuclear data uncertainty analysis results, which were presented in NUREG/CR-7109 and served 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 resolved resonance calculations and performed KENO V.a Monte Carlo method neutron transport calculations.

The same computational method was used to generate the reference FP&MA worth results, which are presented in Table 3.1. The models described in Section 2, including the burned fuel composition information provided in Table 2.4, were used to calculate the reference FP&MA worths. Nominal calculations were performed with all of the nuclides present. Individual calculations were also performed with each of the FP and MA removed, with all FP&MA removed, with all MA removed, and with all FP removed. The group or individual nuclide worth values were then calculated as the change in k_{eff} because of the presence of the nuclide or group of nuclides using the following equation:

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

The results presented in the “All FP&MA,” “All MA,” and “All FP” rows of Table 3.1 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 3.1. Reference FP&MA worth results using SCALE 6.1 and the ENDF/B–VII 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.

4 MCNP RESULTS

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 and -VII.1. FP&MA worth results for these data sets and codes are provided in Table 4.1.

Table 4.1. Variation in calculated FP&MA worths using various codes and data

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	-0.06748	-0.06969	-0.10213	-0.10456
	(% change)[c]	+0.1	+0.7	+0.1	+0.1
MCNP6	ENDF/B-VII	-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] The SCALE nuclear data libraries are described in Section M4 of Ref. **Error! Bookmark not defined.**. The MCNP nuclear data libraries are described in Ref. 4.
[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. 1) and more specifically on the work reported in Section 7.4.3 of Ref. 1. 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 burnup credit 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 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 CSAS5 versus MCNP.

Under prediction 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 1.5% of FP&MA worth bias term. Consequently, variation from the reference values is tolerable, provided the calculated FP&MA worths do not exceed the reference results by more than 1.5% of the FP&MA worths.

These results support the use 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 4.1 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 of the reference worths provided in Table 4.1 appear to exceed the restriction provided in Recommendation 4 of ISG-8, Rev. 3, that the credited minor actinide and fission product 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 0.10442 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.

Detailed FP&MA worth results for each code and nuclear data combination are provided in Appendix C.

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 FP&MA worth results are not affected by which MCNP version is used. Review of the data in Table 4.1 reveals that FP&MA worths calculated using MCNP5 and MCNP6 using the same nuclear data set are statistically the same (i.e. 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.

5 SUMMARY AND 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, insufficient critical experiment data is available to support use of the conventional validation approach for fission products and minor actinides in BUC criticality safety evaluations for PWR SNF casks. Work documented in NUREG/CR-7109 (Ref. 1) supports the use of nuclear data uncertainty to provide a bounding estimate of the potential bias associated with taking credit for FP&MA in burnup credit criticality analyses. Based on that work, NRC Division of Spent Fuel Management 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 report provides justification for use of 1.5% of the FP&MA worth as a bias in BUC criticality safety evaluations using MCNP with the ENDF/B–V, –VI, –VII 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.

Section 2 of this report provides complete descriptions of the reference cask model, including dimensions, materials, and SNF compositions. These models were used to calculate FP&MA worths using MCNP and its ENDF/B–V, –VI, –VII and –VII.1 data. These FP&MA worths are compared with the reference SCALE results to show that they are similar.

The results generated using MCNP with multiple sets of nuclear data are presented in Section 4. 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, and 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 or –VII.1 data is recommended.

6 REFERENCES

1. J. M. Scaglione, et al., *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Analyses – Criticality (k_{eff}) Predictions*, NUREG/CR-7109 (ORNL/TM-2011/514), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn., April 2012.

2. 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*, U.S. Nuclear Regulatory Commission, September 27, 2012.

3. *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, Tenn. June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

4. MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, LA-UR-03-1987, Los Alamos National Laboratory, Los Alamos, N.M., February 1, 2008. Available from Radiation Safety Information and Computational Center at Oak Ridge National Laboratory as CCC-740.

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, Tenn., October 2001.

6. C. D. Harmon II, et al., *Criticality Calculations with MCNP: A Primer*, LA-12827-M, Los Alamos National Laboratory, Los Alamos, N.M., 1994.

7. J. J. Duderstadt and L. J. Hamilton, *Nuclear Reactor Analysis*, John Wiley & Sons, 1976.

8. *Boral–The Proven Neutron Absorber*, AAR Advanced Structures, Livonia, Michigan.

9. *Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*, NUREG-1536, Revision 1, U.S. Nuclear Regulatory Commission, Washington, D.C., July 2010.

10. RW-859 Nuclear Fuel Data, Energy Information Administration, Washington, D.C., October 2004.

11. J. C. Wagner, M. D. DeHart, and C. V. Parks, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*, NUREG/CR-6801 (ORNL/TM-2001/273) prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn., March 2003.

APPENDIX A. SCALE 6.1 CSAS5 MODEL

This appendix includes a SCALE 6.1 CSAS5 model and spent nuclear fuel composition information in the SCALE input format for fuel burned to 20 GWd/MTU, following a 5-year post-irradiation cooling time. Composition inputs for other burnup/cooling time combinations are provided following this CSAS5 input file.

```
=csas5
gbc-32 FP&MA reference worth model
v7-238
read comp

' 3.14 wt%, 20 gwd/mtu, 5 year cooling time
  u-234       101   0  7.3901E-08   293.0 end
  u-235       101   0  3.6046E-04   293.0 end
  u-236       101   0  7.0106E-05   293.0 end
  u-238       101   0  2.2345E-02   293.0 end
 np-237       101   0  5.5048E-06   293.0 end
 pu-238       101   0  1.0223E-06   293.0 end
 pu-239       101   0  1.3719E-04   293.0 end
 pu-240       101   0  3.9569E-05   293.0 end
 pu-241       101   0  1.6819E-05   293.0 end
 pu-242       101   0  3.8879E-06   293.0 end
 am-241       101   0  4.8440E-06   293.0 end
 am-243       101   0  5.1541E-07   293.0 end
 mo-95        101   0  2.8435E-05   293.0 end
 tc-99        101   0  2.9026E-05   293.0 end
 ru-101       101   0  2.6347E-05   293.0 end
 rh-103       101   0  1.8264E-05   293.0 end
 ag-109       101   0  2.1375E-06   293.0 end
 cs-133       101   0  3.0552E-05   293.0 end
 nd-143       101   0  2.2819E-05   293.0 end
 nd-145       101   0  1.6695E-05   293.0 end
 sm-147       101   0  5.6742E-06   293.0 end
 sm-149       101   0  1.7982E-07   293.0 end
 sm-150       101   0  6.4540E-06   293.0 end
 sm-151       101   0  4.1578E-07   293.0 end
 eu-151       101   0  1.6613E-08   293.0 end
 sm-152       101   0  2.6638E-06   293.0 end
 eu-153       101   0  2.2518E-06   293.0 end
 gd-155       101   0  7.0661E-08   293.0 end
  o-16        101   0  4.6943E-02   293.0 end

 zr             2   0  0.0423       293.0 end
 o-16           3   0  0.03337      293.0 end
 h-1            3   0  0.06674      293.0 end
 o-16           4   0  0.03337      293.0 end
 h-1            4   0  0.06674      293.0 end
 cr             5   0  0.01743      293.0 end
 mn             5   0  0.00174      293.0 end
 fe             5   0  0.05936      293.0 end
 ni             5   0  0.00772      293.0 end
 b-10           6   0  6.5794E-03   293.0 end
 b-11           6   0  2.7260E-02   293.0 end
 c              6   0  8.4547E-03   293.0 end
```

```
 al-27          6   0  4.1795E-02   293.0 end
 cr             7   0  0.01743      293.0 end
 mn             7   0  0.00174      293.0 end
 fe             7   0  0.05936      293.0 end
 ni             7   0  0.00772      293.0 end
 al             8   0  0.0602       293.0 end
 o-16           9   0  0.03337      293.0 end
 h-1            9   0  0.06674      293.0 end
 o-16          10   0  0.03337      293.0 end
 h-1           10   0  0.06674      293.0 end
 zr            11   0  0.0423       293.0 end
end comp

read celldata
latticecell  squarepitch
  fueld=   0.7844 101
  gapd=    0.8001   4
  cladd=   0.9144   2
  pitch=   1.2598   3  end
end celldata

read param
 res=1000000 tme=10000 gen=10100 nsk=100 npg=10000 sig=0.0001
 htm=no
end param

read geom

unit 1
  cylinder 101 1 0.3922                           365.76 0
  cylinder   4 1 0.40005                          365.76 0
  cylinder   2 1 0.4572                           365.76 0
  cuboid     3 1 4p0.6299                         365.76 0
'
' Guide Thimble/Instrument Tube (assumed to be same)
unit 2
  cylinder 10 1 0.56135                           365.76 0
  cylinder 11 1 0.6020                            365.76 0
  cuboid   10 1 4p0.6299                          365.76 0
'
' Top Half Horizontal Boral Panel
unit 40
  cuboid   8 1 9.525 -9.525 0.0254  0             365.76 0
  cuboid   6 1 9.525 -9.525 0.12827 0             365.76 0
'
' Right-Hand Side Half Vertical Boral Panel
unit 50
  cuboid   8 1 0.0254  0 9.525 -9.525             365.76 0
  cuboid   6 1 0.12827 0 9.525 -9.525             365.76 0
'
' Bottom Half Horizontal Boral Panel
unit 60
  cuboid   8 1 9.525 -9.525 0 -0.0254             365.76 0
  cuboid   6 1 9.525 -9.525 0 -0.12827            365.76 0
'
' Left-Hand Side Half Vertical Boral Panel
unit 70
```

```
  cuboid   8 1 0 -0.0254  9.525 -9.525        365.76 0
  cuboid   6 1 0 -0.12827 9.525 -9.525        365.76 0
'
' Assembly Basket Cell
unit 101
  array    1 -10.7083 -10.7083 0
  cuboid   9 1 4p11                           365.76 0
  cuboid   5 1 4p11.75                        365.76 0
  cuboid   9 1 4p11.87827                     365.76 0
    hole  40   0     11.75 0
    hole  50  11.75   0    0
    hole  60   0    -11.75 0
    hole  70 -11.75   0    0
'
' Top Boral/Basket Plate
unit 110
  cuboid   6 1  9.525  -9.525 0.10287 0       365.76 0
  cuboid   8 1  9.525  -9.525 0.12827 0       365.76 0
  cuboid   9 1 11.75  -11.75  0.12827 0       365.76 0
  cuboid   5 1 11.75  -11.75  0.87827 0       365.76 0
'
' Left-Hand Side Boral/Basket Plate
unit 112
  cuboid   6 1 0 -0.10287  9.525   -9.525     365.76 0
  cuboid   8 1 0 -0.12827  9.525   -9.525     365.76 0
  cuboid   9 1 0 -0.12827 10.9999 -10.9999    365.76 0
  cuboid   5 1 0 -0.87827 10.9999 -10.9999    365.76 0
'
' Right-Hand Side Boral/Basket Plate
unit 113
  cuboid   6 1 0.10287 0  9.525   -9.525      365.76 0
  cuboid   8 1 0.12827 0  9.525   -9.525      365.76 0
  cuboid   9 1 0.12827 0 10.9999 -10.9999     365.76 0
  cuboid   5 1 0.87827 0 10.9999 -10.9999     365.76 0
unit 114
  array 3 -47.51308 0 0
'
' Cask Inner Volume
global unit 200
  array      2  -71.26962 0     0
  zhemicyl+y 9 1 87.5                         395.76 -15
    hole   114         0 47.51309 0
'
' Exterior Half Boral Panels
' Top Plates
    hole   110 -35.63481 71.26964 0
    hole   110 -11.87827 71.26964 0
    hole   110  11.87827 71.26964 0
    hole   110  35.63481 71.26964 0
    hole   110  59.39135 47.51309 0
    hole   110 -59.39135 47.51309 0
' Left-Hand Side Plates
    hole   112 -47.51310 59.39135 0
    hole   112 -71.26964 35.63481 0
    hole   112 -71.26964 11.87827 0
' Right-Hand Side Plates
    hole   113  47.51310 59.39135 0
```

```
    hole   113  71.26964 35.63481 0
    hole   113  71.26964 11.87827 0
' Steel Cask/Overpack
  zhemicyl+y   7 1 107.5                      425.76 -45
'
' Cuboid Surrounding Cask
  cuboid       0 1 108  -108   108 0          425.76 -45
end geom
'
'
' Assembly Type: Westinghouse 17x17 OFA/V5
read array
  ara= 1 nux= 17 nuy= 17 nuz= 1
    fill
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 1
      1 1 1 2 1 1 1 1 1 1 1 1 1 2 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 2 1 1 1 1 1 1 1 1 1 2 1 1 1
      1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
    end fill
  ara=2 nux=6 nuy=2 nuz=1    fill  12r101  end fill
  ara=3 nux=4 nuy=1 nuz=1    fill   4r101  end fill
end array
read bounds
  +xb=specular
  -xb=specular
  +yb=specular
  +zb=void
  -zb=void
' the -yb must be kept at specular because only half of cask is modeled
  -yb=specular
end bounds
end data
end
```

Composition input for fuel burned to 20 GWd/MTU, following a 40-year cooling time.

Replaces composition 101 provided in the above generic burnup credit–32 (GBC-32) cask model input file.

```
' 3.14 wt%, 20 gwd/mtu, 40 year cooling time
  u-234       101   0  3.2108E-07   293.0 end
  u-235       101   0  3.6060E-04   293.0 end
  u-236       101   0  7.0252E-05   293.0 end
  u-238       101   0  2.2345E-02   293.0 end
 np-237       101   0  6.2485E-06   293.0 end
 pu-238       101   0  7.7547E-07   293.0 end
 pu-239       101   0  1.3705E-04   293.0 end
 pu-240       101   0  3.9477E-05   293.0 end
 pu-241       101   0  3.0797E-06   293.0 end
 pu-242       101   0  3.8877E-06   293.0 end
 am-241       101   0  1.7840E-05   293.0 end
 am-243       101   0  5.1371E-07   293.0 end
 mo-95        101   0  2.8435E-05   293.0 end
 tc-99        101   0  2.9023E-05   293.0 end
 ru-101       101   0  2.6347E-05   293.0 end
 rh-103       101   0  1.8264E-05   293.0 end
 ag-109       101   0  2.1375E-06   293.0 end
 cs-133       101   0  3.0552E-05   293.0 end
 nd-143       101   0  2.2819E-05   293.0 end
 nd-145       101   0  1.6695E-05   293.0 end
 sm-147       101   0  7.4723E-06   293.0 end
 sm-149       101   0  1.7982E-07   293.0 end
 sm-150       101   0  6.4540E-06   293.0 end
 sm-151       101   0  3.1754E-07   293.0 end
 eu-151       101   0  1.1485E-07   293.0 end
 sm-152       101   0  2.6639E-06   293.0 end
 eu-153       101   0  2.2518E-06   293.0 end
 gd-155       101   0  1.3548E-07   293.0 end
  o-16        101   0  4.6943E-02   293.0 end
```

Composition input for fuel burned to 40 GWd/MTU, following a 5-year cooling time.

Replaces composition 101 provided in the above GBC-32 input file.

```
' 4.68 wt%, 40 gwd/mtu, 5 year cooling time
  u-234       101   0  2.6245E-07   293.0 end
  u-235       101   0  3.6741E-04   293.0 end
  u-236       101   0  1.3508E-04   293.0 end
  u-238       101   0  2.1683E-02   293.0 end
 np-237       101   0  1.3907E-05   293.0 end
 pu-238       101   0  4.5238E-06   293.0 end
 pu-239       101   0  1.7413E-04   293.0 end
 pu-240       101   0  6.3031E-05   293.0 end
 pu-241       101   0  3.2475E-05   293.0 end
 pu-242       101   0  1.1666E-05   293.0 end
 am-241       101   0  9.7110E-06   293.0 end
 am-243       101   0  2.5940E-06   293.0 end
 mo-95        101   0  5.4717E-05   293.0 end
 tc-99        101   0  5.5148E-05   293.0 end
 ru-101       101   0  5.1618E-05   293.0 end
 rh-103       101   0  3.1900E-05   293.0 end
 ag-109       101   0  4.3961E-06   293.0 end
 cs-133       101   0  5.7163E-05   293.0 end
 nd-143       101   0  4.1309E-05   293.0 end
 nd-145       101   0  3.1395E-05   293.0 end
 sm-147       101   0  9.3467E-06   293.0 end
 sm-149       101   0  2.2875E-07   293.0 end
 sm-150       101   0  1.3517E-05   293.0 end
 sm-151       101   0  6.4777E-07   293.0 end
 eu-151       101   0  2.5993E-08   293.0 end
 sm-152       101   0  4.2868E-06   293.0 end
 eu-153       101   0  4.9812E-06   293.0 end
 gd-155       101   0  1.7686E-07   293.0 end
  o-16        101   0  4.6949E-02   293.0 end
```

Composition input for fuel burned to 40 GWd/MTU, following a 40-year cooling time.

Replaces composition 101 provided in the above GBC-32 input file.

```
' 4.68 wt%, 40 gwd/mtu, 40 year cooling time
  u-234       101   0  1.3559E-06   293.0 end
  u-235       101   0  3.6758E-04   293.0 end
  u-236       101   0  1.3532E-04   293.0 end
  u-238       101   0  2.1683E-02   293.0 end
 np-237       101   0  1.5362E-05   293.0 end
 pu-238       101   0  3.4316E-06   293.0 end
 pu-239       101   0  1.7396E-04   293.0 end
 pu-240       101   0  6.3301E-05   293.0 end
 pu-241       101   0  5.9464E-06   293.0 end
 pu-242       101   0  1.1665E-05   293.0 end
 am-241       101   0  3.4784E-05   293.0 end
 am-243       101   0  2.5854E-06   293.0 end
 mo-95        101   0  5.4717E-05   293.0 end
 tc-99        101   0  5.5142E-05   293.0 end
 ru-101       101   0  5.1618E-05   293.0 end
 rh-103       101   0  3.1900E-05   293.0 end
 ag-109       101   0  4.3961E-06   293.0 end
 cs-133       101   0  5.7163E-05   293.0 end
 nd-143       101   0  4.1309E-05   293.0 end
 nd-145       101   0  3.1395E-05   293.0 end
 sm-147       101   0  1.1950E-05   293.0 end
 sm-149       101   0  2.2875E-07   293.0 end
 sm-150       101   0  1.3517E-05   293.0 end
 sm-151       101   0  4.9471E-07   293.0 end
 eu-151       101   0  1.7905E-07   293.0 end
 sm-152       101   0  4.2871E-06   293.0 end
 eu-153       101   0  4.9812E-06   293.0 end
 gd-155       101   0  3.3841E-07   293.0 end
  o-16        101   0  4.6949E-02   293.0 end
```

APPENDIX B. MCNP MODEL

This appendix includes a MCNP model and spent nuclear fuel composition information in the MCNP input format for fuel burned to 20 GWd/MTU, following a 5-year post-irradiation cooling time. Composition inputs for other burnup/cooling time combinations are provided following this MCNP input file.

```
GBC-32, FP&MA reference worth model
c
c *****************************************************
c *     GBC-32 cask  (NUREG/CR-6747)
c *     Assembly Type: Westinghouse 17x17 OFA/V5
c *
c *****************************************************
c    ---------- ASSEMBLY ----------
c fuel rod
1     1   7.0120e-02 -1                                 u=2  $ fuel
2     3   0.10011     1   -2                            u=2  $ gap
3     2   0.0423      2   -3                            u=2  $ Zr Clad
4     3   0.10011          3                            u=2  $ water in fuel
cell
c guide tube
5     3   0.10011    -4:5                               u=3  $ water in guide
tubes
6     2   0.0423      4   -5                            u=3  $ guide tubes
c assembly
7     3   0.10011    -6    7   -8    9                  u=1
      lat=1 fill= -9:9    -9:9   0:0
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 2 2 2 1
      1 2 2 2 3 2 2 2 2 2 2 2 2 2 3 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 3 2 2 2 2 2 2 2 2 2 3 2 2 2 1
      1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
c
c
c    ---------- STANDARD BASKET ----------
c
c     Universe 4 = standard symetric cell with assembly
10    0               20  -21   22  -23  fill=1          u=4
11    4   0.08625    30  -31   32  -33  (-20:21:-22:23) u=4
c     right
```

```
20    3   0.10011    31                 #21 #22         u=4 $ Water
21    7   0.0602     31  -41   12  -13                  u=4 $ Al Boral Clad
22    5   0.084089   41        12  -13                  u=4 $ Boral Center
c     left
30    3   0.10011   -30                 #31 #32         u=4 $ Water
31    7   0.0602    -30   40   12  -13                  u=4 $ Al Boral Clad
32    5   0.084089  -40        12  -13                  u=4 $ Boral Center
c     top
40    3   0.10011    33        30  -31  #41 #42         u=4 $ Water
41    7   0.0602     33  -43   10  -11                  u=4 $ Al Boral Clad
42    5   0.084089   43        10  -11                  u=4 $ Boral Center
c     bottom
50    3   0.10011   -32        30  -31  #51 #52         u=4 $ Water
51    7   0.0602    -32   42   10  -11                  u=4 $ Al Boral Clad
52    5   0.084089  -42        10  -11                  u=4 $ Boral Center
c
c
c    ---------- PERIPHERY BASKET/POISON PLATES ----------
c
c     Universe 5 = top cell, which has boral on bottom
100    3   0.10011    22                                u=5
101    3   0.10011   -22                 #102 #103 #104 u=5
102    4   0.08625    30  -31  -22   32                 u=5
103    7   0.0602    -32   42   10  -11                 u=5 $ Al Boral Clad
104    5   0.084089  -42        10  -11                 u=5 $ Boral Center
c
c     Universe 6 = bottom cell, which has boral on top
110    3   0.10011   -23                                u=6
111    3   0.10011    23                 #112 #113 #114 u=6
112    4   0.08625    30  -31   23  -33                 u=6
113    7   0.0602     33  -43   10  -11                 u=6 $ Al Boral Clad
114    5   0.084089   43        10  -11                 u=6 $ Boral Center
c
c     Universe 7 = right-hand side cell, which has boral on left-hand side
120    3   0.10011    20                                u=7
121    3   0.10011   -20                 #122 #123 #124 u=7
122    4   0.08625    22  -23  -20   30                 u=7
123    7   0.0602    -30   40   12  -13                 u=7 $ Al Boral Clad
124    5   0.084089  -40        12  -13                 u=7 $ Boral Center
c
c     Universe 8 = left-hand side cell, which has boral on right-hand side
130    3   0.10011   -21                                u=8
131    3   0.10011    21                 #132 #133 #134 u=8
132    4   0.08625    22  -23   21  -31                 u=8
133    7   0.0602     31  -41   12  -13                 u=8 $ Al Boral Clad
134    5   0.084089   41        12  -13                 u=8 $ Boral Center
c
c
c    ---------- BASKET CORNERS ----------
c
c     Universe 10 = top-left corner cell, which has boral on bottom-right
150    3   0.10011   -21   22                           u=10
151    4   0.08625   -31   32  (21:-22)    #152 #153    u=10
152    3   0.10011    21  -31  23   -53                 u=10
153    3   0.10011    50  -30  32   -22                 u=10
c     right
155    3   0.10011    31                   #156 #157    u=10 $ Water
```

```
156    7   0.0602     31  -41   12  -13                  u=10 $ Al Boral Clad
157    5   0.084089   41        12  -13                  u=10 $ Boral Center
c     bottom
160    3   0.10011   -32            -31    #161 #162    u=10 $ Water
161    7   0.0602    -32   42   10  -11                  u=10 $ Al Boral Clad
162    5   0.084089  -42        10  -11                  u=10 $ Boral Center
c
c
c     Universe 11 = top-right corner cell, which has boral on bottom-left
170    3   0.10011    20   22                            u=11
171    4   0.08625    30   32 (-20:-22)   #172 #173     u=11
172    3   0.10011    30  -20 23 -53                     u=11
173    3   0.10011    31  -51 32 -22                     u=11
c     left
175    3   0.10011   -30                   #176 #177    u=11 $ Water
176    7   0.0602    -30   40   12  -13                  u=11 $ Al Boral Clad
177    5   0.084089  -40        12  -13                  u=11 $ Boral Center
c     bottom
180    3   0.10011   -32             30    #181 #182    u=11 $ Water
181    7   0.0602    -32   42   10  -11                  u=11 $ Al Boral Clad
182    5   0.084089  -42        10  -11                  u=11 $ Boral Center
c
c
c     Universe 12 = bottom-left corner cell, which has boral on top-right
190    3   0.10011   -21  -23                            u=12
191    4   0.08625   -31  -33  (21:23)     #192 #193    u=12
192    3   0.10011    50  -30  23 -33                    u=12
193    3   0.10011    21  -31  52 -22                    u=12
c     right
195    3   0.10011    31                   #196 #197    u=12 $ Water
196    7   0.0602     31  -41   12  -13                  u=12 $ Al Boral Clad
197    5   0.084089   41        12  -13                  u=12 $ Boral Center
c     top
200    3   0.10011    33            -31    #201 #202    u=12 $ Water
201    7   0.0602     33  -43   10  -11                  u=12 $ Al Boral Clad
202    5   0.084089   43        10  -11                  u=12 $ Boral Center
c
c
c     Universe 13 = bottom-right corner cell, which has boral on top-left
210    3   0.10011    20  -23                            u=13
211    4   0.08625    30  -33 (-20:23)     #212 #213    u=13
212    3   0.10011    31  -51 23 -33                     u=13
213    3   0.10011    30  -20 52 -22                     u=13
c     left
215    3   0.10011   -30                   #216 #217    u=13 $ Water
216    7   0.0602    -30   40   12  -13                  u=13 $ Al Boral Clad
217    5   0.084089  -40        12  -13                  u=13 $ Boral Center
c     top
220    3   0.10011    33             30    #221 #222    u=13 $ Water
221    7   0.0602     33  -43   10  -11                  u=13 $ Al Boral Clad
222    5   0.084089   43        10  -11                  u=13 $ Boral Center
c
c
c    ---------- CASK ----------
c
c     Cask Loading Pattern
500   3  0.10011 50 -51 52 -53  u=20 lat=1
```

```
                fill=-5:4 -5:4 0:0
                20  20  20  20  20  20  20  20  20  20
                20  20  20   5   5   5   5  20  20  20
                20  20  11   4   4   4   4  10  20  20
                20   7   4   4   4   4   4   4   8  20
                20   7   4   4   4   4   4   4   8  20
                20   7   4   4   4   4   4   4   8  20
                20   7   4   4   4   4   4   4   8  20
                20  20  13   4   4   4   4  12  20  20
                20  20  20   6   6   6   6  20  20  20
                20  20  20  20  20  20  20  20  20  20
c         rhs                                          lhs
c
600   0          -200  101 -102   fill=20  (-11.87827  -11.87827 0.0)
601   3  0.10011 -200  100 -101       $ water above baseplate and below fuel
602   6  0.08625 -201   99 -100       $ steel baseplate
603   3  0.10011 -200  102 -103       $ water above fuel and below cask lid
604   6  0.08625 -201  103 -104       $ steel cask lid
605   6  0.08625  200 -201  100 -103  $ cask
700   0           300 -301  302 -303 99 -104 201 $ enclosing void
701   0          (-300:301:-302:303:-99:104)      $ outside universe

1     cz         0.3922      $ Pellet O.R.
2     cz         0.40005     $ Clad I.R.
3     cz         0.4572      $ Clad O.R.
4     cz         0.56135     $ Water rod I.R.
5     cz         0.6020      $ Water rod O.R.
6     px         0.6299      $ Fuel rod half-pitch
7     px        -0.6299      $         "
8     py         0.6299      $         "
9     py        -0.6299      $         "
10    px        -9.525       $ boral plate half-width
11    px         9.525       $         "
12    py        -9.525       $         "
13    py         9.525       $         "
20    px       -11.0000      $ SS Box cell I.D., 8.6614"
21    px        11.0000      $         "
22    py       -11.0000      $         "
23    py        11.0000      $         "
30    px       -11.7500      $ SS Box cell O.D., thkns=0.2953"
31    px        11.7500      $         "
32    py       -11.7500      $         "
33    py        11.7500      $         "
40    px       -11.7754      $ Al clad thickness outside Boral
41    px        11.7754      $         "
42    py       -11.7754      $         "
43    py        11.7754      $         "
50    px       -11.87825     $ cell half-pitch (to center of Boral plate)
51    px        11.87825     $         "
52    py       -11.87825     $         "
53    py        11.87825     $         "
c
c   axial planes
99     pz      -45.          $ Bottom of Cask Baseplate
100    pz      -15.          $ Top of Cask Baseplate
101    pz        0.          $ Bottom of fuel
102    pz      365.76        $ Top of fuel
```

```
103    pz      395.76        $ Bottom of Cask Lid
104    pz      425.76        $ Top of Cask Lid
c
200    cz       87.5         $ Cask body inner radius
201    cz      107.5         $ Cask body outer radius
c
*300   px     -108.0         $ Outer Enclosing Box
*301   px      108.0
*302   py     -108.0
*303   py      108.0

imp:n    1  88r  0
kcode    10000  0.95  100  10000 0    0    200  1
c        NSRCK  RKK   IKZ  KCT   MSRK KNRM MRKP KC8
c
c
c   Initial Source = Uniform throughout assemblies
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1  -2 1.2895
c
si3  h 0 365.76
sp3  0 1
c
c
si4  s      12 13 14 15
         11 12 13 14 15 16
         11 12 13 14 15 16
         11 12 13 14 15 16
         11 12 13 14 15 16
            12 13 14 15
sp4  1 31r
c
ds5  s      26 26 26 26
         25 25 25 25 25 25
         24 24 24 24 24 24
         23 23 23 23 23 23
         22 22 22 22 22 22
            21 21 21 21
c
si11  -66.3500 -44.9327
si12  -44.0935 -22.6762
si13  -21.8369  -0.4196
si14    0.4196  21.8369
si15   22.6762  44.0935
si16   44.9327  66.3500
c
si21  -66.3500 -44.9327
si22  -44.0935 -22.6762
si23  -21.8369  -0.4196
si24    0.4196  21.8369
si25   22.6762  44.0935
si26   44.9327  66.3500
c
sp11  0 1
sp12  0 1
```

```
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
c
c
c
c     Material Cards
c
c  Fuel  20 GWd/MTU & 5 year cooling time
c          (total = 7.0120e-02)
m1      92234.70c   7.3901e-08
        92235.70c   3.6046e-04
        92236.70c   7.0106e-05
        92238.70c   2.2345e-02
        93237.70c   5.5048e-06
        94238.70c   1.0223e-06
        94239.70c   1.3719e-04
        94240.70c   3.9569e-05
        94241.70c   1.6819e-05
        94242.70c   3.8879e-06
        95241.70c   4.8440e-06
        95243.70c   5.1541e-07
        42095.70c   2.8435e-05
        43099.70c   2.9026e-05
        44101.70c   2.6347e-05
        45103.70c   1.8264e-05
        47109.70c   2.1375e-06
        55133.70c   3.0552e-05
        60143.70c   2.2819e-05
        60145.70c   1.6695e-05
        62147.70c   5.6742e-06
        62149.70c   1.7982e-07
        62150.70c   6.4540e-06
        62151.70c   4.1578e-07
        63151.70c   1.6613e-08
        62152.70c   2.6638e-06
        63153.70c   2.2518e-06
        64155.70c   7.0661e-08
         8016.70c   4.6943e-02
c
c  Zr clad
m2      40090.70c   2.1763e-02
        40091.70c   4.7461e-03
        40092.70c   7.2545e-03
        40094.70c   7.3517e-03
        40096.70c   1.1844e-03
c   Water
m3       8016.70c   3.3370e-02
         1001.70c   6.6740e-02
mt3      lwtr.10t
```

```
c
c       Basket Material - Stainless Steel Type 304
m4      24050.70c   7.5821e-04
        24052.70c   1.4605e-02
        24053.70c   1.6559e-03
        24054.70c   4.1135e-04
        25055.70c   1.7400e-03
        26054.70c   3.4726e-03
        26056.70c   5.4463e-02
        26057.70c   1.2584e-03
        26058.70c   1.6621e-04
        28058.70c   5.2558e-03
        28060.70c   2.0242e-03
        28061.70c   8.8008e-05
        28062.70c   2.8024e-04
        28064.70c   7.1796e-05
c
c       Boral Central Section @ 0.0225 g/cmsq
m5       5010.70c   6.5794e-03
         5011.70c   2.7260e-02
         6000.70c   8.4547e-03
        13027.70c   4.1795e-02
c
c       Cask Material - Stainless Steel Type 304
m6      24050.70c   7.5821e-04
        24052.70c   1.4605e-02
        24053.70c   1.6559e-03
        24054.70c   4.1135e-04
        25055.70c   1.7400e-03
        26054.70c   3.4726e-03
        26056.70c   5.4463e-02
        26057.70c   1.2584e-03
        26058.70c   1.6621e-04
        28058.70c   5.2558e-03
        28060.70c   2.0242e-03
        28061.70c   8.8008e-05
        28062.70c   2.8024e-04
        28064.70c   7.1796e-05
c
c       Aluminum Clad for Boral
m7      13027.70c   6.0200e-02
c
prdmp  j   -60   j   2
```

Composition input for fuel burned to 20 GWd/MTU, following a 40-year cooling time.

The total number density in cell 1 is changed to 7.0122e-02

Replace composition m1 provided in the above generic burnup credit–32 (GBC-32) cask model input file.

```
c  Fuel  20 GWd/MTU & 40 year cooling time
c           (total = 7.0122e-02)
m1       92234.70c   3.2108e-07
         92235.70c   3.6060e-04
         92236.70c   7.0252e-05
         92238.70c   2.2345e-02
         93237.70c   6.2485e-06
         94238.70c   7.7547e-07
         94239.70c   1.3705e-04
         94240.70c   3.9477e-05
         94241.70c   3.0797e-06
         94242.70c   3.8877e-06
         95241.70c   1.7840e-05
         95243.70c   5.1371e-07
         42095.70c   2.8435e-05
         43099.70c   2.9023e-05
         44101.70c   2.6347e-05
         45103.70c   1.8264e-05
         47109.70c   2.1375e-06
         55133.70c   3.0552e-05
         60143.70c   2.2819e-05
         60145.70c   1.6695e-05
         62147.70c   7.4723e-06
         62149.70c   1.7982e-07
         62150.70c   6.4540e-06
         62151.70c   3.1754e-07
         63151.70c   1.1485e-07
         62152.70c   2.6639e-06
         63153.70c   2.2518e-06
         64155.70c   1.3548e-07
          8016.70c   4.6943e-02
```

Composition input for fuel burned to 40 GWd/MTU, following a 5-year cooling time.

The total number density in cell 1 is changed to 6.9808e-02

Replace composition m1 provided in the above GBC-32 input file.

```
c  Fuel  40 GWd/MTU & 5 year cooling time
c          (total = 6.9808e-02)
m1      92234.70c   2.6245e-07
        92235.70c   3.6741e-04
        92236.70c   1.3508e-04
        92238.70c   2.1683e-02
        93237.70c   1.3907e-05
        94238.70c   4.5238e-06
        94239.70c   1.7413e-04
        94240.70c   6.3031e-05
        94241.70c   3.2475e-05
        94242.70c   1.1666e-05
        95241.70c   9.7110e-06
        95243.70c   2.5940e-06
        42095.70c   5.4717e-05
        43099.70c   5.5148e-05
        44101.70c   5.1618e-05
        45103.70c   3.1900e-05
        47109.70c   4.3961e-06
        55133.70c   5.7163e-05
        60143.70c   4.1309e-05
        60145.70c   3.1395e-05
        62147.70c   9.3467e-06
        62149.70c   2.2875e-07
        62150.70c   1.3517e-05
        62151.70c   6.4777e-07
        63151.70c   2.5993e-08
        62152.70c   4.2868e-06
        63153.70c   4.9812e-06
        64155.70c   1.7686e-07
         8016.70c   4.6949e-02
```

Composition input for fuel burned to 40 GWd/MTU, following a 40-year cooling time.

The total number density in cell 1 is changed to 6.9811e-02

Replace composition m1 provided in the above GBC-32 input file.

```
c  Fuel  40 GWd/MTU & 40 year cooling time
c           (total = 6.9811e-02)
m1       92234.70c   1.3559e-06
         92235.70c   3.6758e-04
         92236.70c   1.3532e-04
         92238.70c   2.1683e-02
         93237.70c   1.5362e-05
         94238.70c   3.4316e-06
         94239.70c   1.7396e-04
         94240.70c   6.3301e-05
         94241.70c   5.9464e-06
         94242.70c   1.1665e-05
         95241.70c   3.4784e-05
         95243.70c   2.5854e-06
         42095.70c   5.4717e-05
         43099.70c   5.5142e-05
         44101.70c   5.1618e-05
         45103.70c   3.1900e-05
         47109.70c   4.3961e-06
         55133.70c   5.7163e-05
         60143.70c   4.1309e-05
         60145.70c   3.1395e-05
         62147.70c   1.1950e-05
         62149.70c   2.2875e-07
         62150.70c   1.3517e-05
         62151.70c   4.9471e-07
         63151.70c   1.7905e-07
         62152.70c   4.2871e-06
         63153.70c   4.9812e-06
         64155.70c   3.3841e-07
          8016.70c   4.6949e-02
```

APPENDIX C. DETAILED FP&MA WORTH RESULTS FOR EACH CODE AND NUCLEAR DATA COMBINATION

This appendix provides detailed fission products and minor actinides (FP&MA) worth results for each code and nuclear data combination. Results for SCALE 6.1 using the ENDF/B–VII 238 group library are provided in Table 3.1.

Table C.1. FP&MA worths using SCALE 6.1 and ENDF/B–VII continuous-energy data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06720	-0.06945	-0.10149	-0.10399
All MA	-0.00609	-0.00657	-0.01120	-0.01096
All FP	-0.06018	-0.06255	-0.08745	-0.09037
^{236}U	-0.00367	-0.00390	-0.00544	-0.00512
^{237}Np	-0.00209	-0.00249	-0.00435	-0.00472
^{243}Am	-0.00021	-0.00058	-0.00094	-0.00099
^{95}Mo	-0.00120	-0.00135	-0.00160	-0.00170
^{99}Tc	-0.00231	-0.00270	-0.00384	-0.00365
^{101}Ru	-0.00073	-0.00121	-0.00146	-0.00134
^{103}Rh	-0.00695	-0.00707	-0.01049	-0.00968
^{109}Ag	-0.00074	-0.00123	-0.00145	-0.00128
^{133}Cs	-0.00334	-0.00352	-0.00521	-0.00497
^{143}Nd	-0.00866	-0.00898	-0.01299	-0.01249
^{145}Nd	-0.00165	-0.00207	-0.00279	-0.00293
^{147}Sm	-0.00128	-0.00212	-0.00199	-0.00239
^{149}Sm	-0.01555	-0.01532	-0.01597	-0.01532
^{150}Sm	-0.00071	-0.00114	-0.00128	-0.00123
^{151}Sm	-0.00584	-0.00465	-0.00701	-0.00522
^{152}Sm	-0.00198	-0.00256	-0.00279	-0.00284
^{151}Eu	-0.00004	-0.00156	0.00002	-0.00150
^{153}Eu	-0.00165	-0.00171	-0.00270	-0.00288
^{155}Gd	-0.00357	-0.00712	-0.00667	-0.01294

[a] Monte Carlo uncertainty is 0.00014 Δk for FP&MA worths.

Table C.2. FP&MA worths using SCALE 6.1 and ENDF/B–VI 238 energy group data

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
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06755	-0.06888	-0.10152	-0.10374
All MA	-0.00673	-0.00618	-0.01158	-0.01087
All FP	-0.06022	-0.06198	-0.08743	-0.09012
^{236}U	-0.00396	-0.00345	-0.00570	-0.00507
^{237}Np	-0.00225	-0.00238	-0.00471	-0.00471
^{243}Am	-0.00023	-0.00007	-0.00098	-0.00080
^{95}Mo	-0.00132	-0.00100	-0.00179	-0.00164
^{99}Tc	-0.00211	-0.00198	-0.00336	-0.00321
^{101}Ru	-0.00069	-0.00078	-0.00141	-0.00111
^{103}Rh	-0.00704	-0.00658	-0.01029	-0.00928
^{109}Ag	-0.00109	-0.00071	-0.00154	-0.00157
^{133}Cs	-0.00341	-0.00296	-0.00531	-0.00487
^{143}Nd	-0.00912	-0.00870	-0.01319	-0.01229
^{145}Nd	-0.00183	-0.00161	-0.00271	-0.00270
^{147}Sm	-0.00143	-0.00155	-0.00185	-0.00252
^{149}Sm	-0.01655	-0.01574	-0.01688	-0.01619
^{150}Sm	-0.00099	-0.00050	-0.00148	-0.00126
^{151}Sm	-0.00629	-0.00419	-0.00723	-0.00534
^{152}Sm	-0.00222	-0.00190	-0.00309	-0.00272
^{151}Eu	-0.00024	-0.00105	-0.00018	-0.00134
^{153}Eu	-0.00170	-0.00123	-0.00286	-0.00270
^{155}Gd	-0.00372	-0.00650	-0.00730	-0.01302

[a] Monte Carlo uncertainty is 0.00014 Δk for FP&MA worths.

Table C.3. FP&MA worths using SCALE 6.1 and ENDF/B–V 238 energy group data

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
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06585	-0.06808	-0.09966	-0.10229
All MA	-0.00615	-0.00632	-0.01159	-0.01109
All FP	-0.05856	-0.06113	-0.08544	-0.08835
^{236}U	-0.00349	-0.00375	-0.00545	-0.00518
^{237}Np	-0.00197	-0.00247	-0.00484	-0.00496
^{243}Am	-0.00016	-0.00049	-0.00110	-0.00111
^{95}Mo	-0.00114	-0.00116	-0.00182	-0.00176
^{99}Tc	-0.00204	-0.00221	-0.00376	-0.00350
^{101}Ru	-0.00056	-0.00077	-0.00118	-0.00137
^{103}Rh	-0.00678	-0.00668	-0.01041	-0.00959
^{109}Ag	-0.00067	-0.00108	-0.00171	-0.00138
^{133}Cs	-0.00272	-0.00299	-0.00463	-0.00450
^{143}Nd	-0.00885	-0.00886	-0.01305	-0.01268
^{145}Nd	-0.00149	-0.00170	-0.00269	-0.00257
^{147}Sm	-0.00134	-0.00186	-0.00210	-0.00270
^{149}Sm	-0.01547	-0.01538	-0.01634	-0.01567
^{150}Sm	-0.00077	-0.00089	-0.00173	-0.00159
^{151}Sm	-0.00552	-0.00445	-0.00700	-0.00537
^{152}Sm	-0.00186	-0.00215	-0.00323	-0.00305
^{151}Eu	0.00019	-0.00146	-0.00026	-0.00176
^{153}Eu	-0.00118	-0.00145	-0.00290	-0.00271
^{155}Gd	-0.00316	-0.00662	-0.00709	-0.01287

[a] Monte Carlo uncertainty is 0.00014 Δk for FP&MA worths.

Table C.4. FP&MA worths using SCALE 6.1 and ENDF/B–V 44 energy group data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06635	-0.06836	-0.10000	-0.10288
All MA	-0.00622	-0.00639	-0.01186	-0.01126
All FP	-0.05915	-0.06132	-0.08593	-0.08896
^{236}U	-0.00402	-0.00404	-0.00606	-0.00542
^{237}Np	-0.00221	-0.00249	-0.00489	-0.00480
^{243}Am	-0.00015	-0.00037	-0.00113	-0.00091
^{95}Mo	-0.00131	-0.00131	-0.00204	-0.00158
^{99}Tc	-0.00276	-0.00253	-0.00420	-0.00379
^{101}Ru	-0.00074	-0.00089	-0.00146	-0.00118
^{103}Rh	-0.00699	-0.00674	-0.01041	-0.00923
^{109}Ag	-0.00084	-0.00117	-0.00166	-0.00137
^{133}Cs	-0.00316	-0.00317	-0.00505	-0.00488
^{143}Nd	-0.00897	-0.00882	-0.01332	-0.01286
^{145}Nd	-0.00154	-0.00186	-0.00283	-0.00255
^{147}Sm	-0.00144	-0.00161	-0.00215	-0.00247
^{149}Sm	-0.01571	-0.01557	-0.01668	-0.01569
^{150}Sm	-0.00133	-0.00139	-0.00209	-0.00206
^{151}Sm	-0.00581	-0.00427	-0.00705	-0.00517
^{152}Sm	-0.00205	-0.00187	-0.00309	-0.00256
^{151}Eu	-0.00033	-0.00144	-0.00038	-0.00128
^{153}Eu	-0.00143	-0.00145	-0.00285	-0.00264
^{155}Gd	-0.00359	-0.00687	-0.00696	-0.01301

[a] Monte Carlo uncertainty is 0.00014 Δk for FP&MA worths.

Table C.5. FP&MA worths using MCNP5 (version 1.60) and ENDF/B–VII data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06748	-0.06969	-0.10213	-0.10456
All MA	-0.00618	-0.00619	-0.01127	-0.01073
All FP	-0.06042	-0.06244	-0.08820	-0.09096
^{236}U (92236.70c)	-0.00376	-0.00374	-0.00580	-0.00523
^{237}Np (93237.70c)	-0.00216	-0.00241	-0.00443	-0.00460
^{243}Am (95243.70c)	-0.00002	-0.00038	-0.00117	-0.00104
^{95}Mo (42095.70c)	-0.00101	-0.00105	-0.00176	-0.00164
^{99}Tc (43099.70c)	-0.00256	-0.00259	-0.00388	-0.00365
^{101}Ru (44101.70c)	-0.00076	-0.00093	-0.00135	-0.00142
^{103}Rh (45103.70c)	-0.00688	-0.00675	-0.01026	-0.00944
^{109}Ag (47109.70c)	-0.00087	-0.00080	-0.00160	-0.00139
^{133}Cs (55133.70c)	-0.00334	-0.00328	-0.00546	-0.00511
^{143}Nd (60143.70c)	-0.00877	-0.00887	-0.01331	-0.01259
^{145}Nd (60145.70c)	-0.00176	-0.00187	-0.00282	-0.00276
^{147}Sm (62147.70c)	-0.00137	-0.00191	-0.00214	-0.00240
^{149}Sm (62149.70c)	-0.01570	-0.01530	-0.01637	-0.01564
^{150}Sm (62150.70c)	-0.00081	-0.00097	-0.00151	-0.00136
^{151}Sm (62151.70c)	-0.00612	-0.00461	-0.00734	-0.00555
^{152}Sm (62152.70c)	-0.00211	-0.00220	-0.00298	-0.00285
^{151}Eu (63151.70c)	-0.00015	-0.00142	-0.00020	-0.00158
^{153}Eu (63153.70c)	-0.00167	-0.00156	-0.00270	-0.00271
^{155}Gd (64155.70c)	-0.00354	-0.00689	-0.00723	-0.01301

Note: MCNP = Monte Carlo N-Particle.
[a] Monte Carlo uncertainty is less than 0.00012 Δk for FP&MA worths.

Table C.6. FP&MA worths using MCNP5 (version 1.60) and ENDF/B–VI data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.04529	-0.04918	-0.06910	-0.07487
All MA	-0.00659	-0.00616	-0.01194	-0.01158
All FP	-0.03797	-0.04222	-0.05474	-0.06139
^{236}U (92236.69c)	-0.00403	-0.00400	-0.00611	-0.00516
^{237}Np (93237.69c)	-0.00214	-0.00242	-0.00486	-0.00500
^{243}Am (95243.69c)	-0.00024	-0.00026	-0.00087	-0.00079
^{95}Mo	[b]	[b]	[b]	[b]
^{99}Tc (43099.66c)	-0.00236	-0.00213	-0.00361	-0.00328
^{101}Ru	[b]	[b]	[b]	[b]
^{103}Rh (45103.66c)	-0.00728	-0.00673	-0.01080	-0.01007
^{109}Ag (47109.66c)	-0.00097	-0.00031	-0.00144	-0.00132
^{133}Cs (55133.66c)	-0.00309	-0.00326	-0.00540	-0.00475
^{143}Nd	[b]	[b]	[b]	[b]
^{145}Nd	[b]	[b]	[b]	[b]
^{147}Sm (62147.66c)	-0.00142	-0.00183	-0.00169	-0.00240
^{149}Sm (62149.66c)	-0.01671	-0.01582	-0.01800	-0.01679
^{150}Sm	[b]	[b]	[b]	[b]
^{151}Sm	[b]	[b]	[b]	[b]
^{152}Sm	[b]	[b]	[b]	[b]
^{151}Eu (63151.66c)	-0.00041	-0.00102	-0.00027	-0.00123
^{153}Eu (63153.66c)	-0.00149	-0.00134	-0.00292	-0.00271
^{155}Gd (64155.66c)	-0.00381	-0.00699	-0.00771	-0.01424

Note: MCNP = Monte Carlo N-Particle.
[a] Monte Carlo uncertainty is less than 0.00014 Δk for FP&MA worths.
[b] ^{95}Mo, ^{101}Ru, ^{143}Nd, ^{145}Nd, ^{150}Sm, ^{151}Sm, and ^{152}Sm excluded because ENDF/B–VI data for these nuclides was not distributed with MCNP5 (v1.60).

Table C.7. FP&MA worths using MCNP5 (version 1.60) and ENDF/B–V data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06688	-0.06913	-0.10154	-0.10402
All MA	-0.00678	-0.00696	-0.01303	-0.01238
All FP	-0.05922	-0.06133	-0.08593	-0.08895
^{236}U (92236.51c)	-0.00452	-0.00428	-0.00698	-0.00627
^{237}Np (93237.55c)	-0.00209	-0.00270	-0.00485	-0.00492
^{243}Am (95243.51c)	-0.00018	-0.00046	-0.00122	-0.00091
^{95}Mo (42095.50c)	-0.00095	-0.00130	-0.00215	-0.00155
^{99}Tc (43099.50c)	-0.00198	-0.00233	-0.00370	-0.00339
^{101}Ru (44101.50c)	-0.00051	-0.00072	-0.00129	-0.00130
^{103}Rh (45103.50c)	-0.00702	-0.00692	-0.01042	-0.00939
^{109}Ag (47109.50c)	-0.00057	-0.00096	-0.00152	-0.00171
^{133}Cs (55133.55c)	-0.00295	-0.00312	-0.00513	-0.00496
^{143}Nd (60143.50c)	-0.00891	-0.00897	-0.01318	-0.01253
^{145}Nd (60145.50c)	-0.00143	-0.00146	-0.00267	-0.00265
^{147}Sm (62147.50c)	-0.00120	-0.00162	-0.00213	-0.00241
^{149}Sm (62149.50c)	-0.01580	-0.01557	-0.01623	-0.01588
^{150}Sm (62150.50c)	-0.00092	-0.00093	-0.00160	-0.00122
^{151}Sm (62151.50c)	-0.00532	-0.00436	-0.00688	-0.00502
^{152}Sm (62152.50c)	-0.00201	-0.00218	-0.00317	-0.00311
^{151}Eu (63151.55c)	-0.00011	-0.00120	-0.00022	-0.00142
^{153}Eu (63153.55c)	-0.00128	-0.00141	-0.00263	-0.00261
^{155}Gd (64155.50c)	-0.00340	-0.00672	-0.00709	-0.01304

Note: MCNP = Monte Carlo N-Particle.
[a] Monte Carlo uncertainty is less than 0.00013 Δk for FP&MA worths.

Table C.8. FP&MA worths using MCNP6 (new release) and ENDF/B–V data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06705	-0.06906	-0.10136	-0.10395
All MA	-0.0069	-0.00673	-0.01271	-0.01233
All FP	-0.05915	-0.06108	-0.08581	-0.08898
^{236}U (92236.51c)	-0.00458	-0.00418	-0.00697	-0.0062
^{237}Np (93237.55c)	-0.00211	-0.00233	-0.00477	-0.0051
^{243}Am (95243.51c)	-0.00016	-0.00028	-0.00114	-0.00096
^{95}Mo (42095.50c)	-0.00106	-0.00107	-0.0019	-0.00184
^{99}Tc (43099.50c)	-0.0023	-0.00207	-0.00356	-0.00342
^{101}Ru (44101.50c)	-0.00075	-0.0005	-0.00113	-0.00119
^{103}Rh (45103.50c)	-0.00701	-0.0066	-0.01033	-0.00942
^{109}Ag (47109.50c)	-0.00088	-0.00076	-0.0013	-0.00139
^{133}Cs (55133.55c)	-0.00315	-0.00288	-0.00512	-0.00479
^{143}Nd (60143.50c)	-0.00885	-0.00847	-0.0132	-0.01262
^{145}Nd (60145.50c)	-0.0017	-0.00147	-0.00259	-0.00247
^{147}Sm (62147.50c)	-0.00145	-0.00162	-0.00203	-0.00252
^{149}Sm (62149.50c)	-0.01593	-0.01532	-0.01642	-0.01571
^{150}Sm (62150.50c)	-0.00063	-0.00077	-0.0015	-0.00144
^{151}Sm (62151.50c)	-0.00552	-0.00395	-0.00686	-0.00499
^{152}Sm (62152.50c)	-0.00227	-0.00211	-0.00309	-0.00283
^{151}Eu (63151.55c)	-0.00008	-0.00109	-0.0002	-0.00149
^{153}Eu (63153.55c)	-0.00142	-0.00129	-0.00288	-0.0025
^{155}Gd (64155.50c)	-0.00357	-0.00661	-0.00706	-0.01316

Note: MCNP = Monte Carlo N-Particle.
[a] Monte Carlo uncertainty is less than 0.00013 Δk for FP&MA worths.

Table C.9. FP&MA worths using MCNP6 (new release) and ENDF/B–VI data

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
	FP&MA worths (Δk)[a]			
All FP&MA	-0.04525	-0.04908	-0.0692	-0.07511
All MA	-0.00658	-0.00641	-0.01215	-0.0118
All FP	-0.03828	-0.04205	-0.05505	-0.06144
^{236}U (92236.69c)	-0.00409	-0.00375	-0.00591	-0.00552
^{237}Np (93237.69c)	-0.00225	-0.0023	-0.00495	-0.00527
^{243}Am (95243.69c)	-0.00007	-0.00028	-0.00101	-0.00106
^{99}Tc (43099.66c)	-0.00235	-0.00206	-0.00377	-0.00352
^{103}Rh (45103.66c)	-0.0072	-0.00687	-0.01072	-0.01007
^{109}Ag (47109.66c)	-0.00087	-0.00076	-0.00138	-0.00145
^{133}Cs (55133.66c)	-0.00305	-0.0031	-0.00521	-0.0050
^{147}Sm (62147.66c)	-0.00136	-0.00174	-0.00198	-0.00264
^{149}Sm (62149.66c)	-0.01680	-0.01613	-0.01775	-0.01692
^{151}Eu (63151.66c)	-0.00016	-0.00117	-0.00001	-0.00154
^{153}Eu (63153.66c)	-0.00152	-0.00131	-0.00291	-0.00283
^{155}Gd (64155.66c)	-0.00363	-0.00707	-0.00772	-0.01431

Note: MCNP = Monte Carlo N-Particle.
[a] Monte Carlo uncertainty is less than 0.00013 Δk for FP&MA worths.

Table C.10. FP&MA worths using MCNP6 (new release) and ENDF/B–VII data

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**
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06762	-0.06961	-0.10222	-0.1045
All MA	-0.00606	-0.00608	-0.01126	-0.01089
All FP	-0.06053	-0.06269	-0.08838	-0.09093
^{236}U (92236.70c)	-0.00393	-0.00368	-0.00579	-0.00514
^{237}Np (93237.70c)	-0.0019	-0.00232	-0.00447	-0.00448
^{243}Am (95243.70c)	-0.00021	-0.00032	-0.0011	-0.00081
^{95}Mo (42095.70c)	-0.00115	-0.00116	-0.00204	-0.0016
^{99}Tc (43099.70c)	-0.00253	-0.0025	-0.00403	-0.00351
^{101}Ru (44101.70c)	-0.00063	-0.00075	-0.00152	-0.0012
^{103}Rh (45103.70c)	-0.00677	-0.00664	-0.01045	-0.00929
^{109}Ag (47109.70c)	-0.00084	-0.00085	-0.00182	-0.0014
^{133}Cs (55133.70c)	-0.00332	-0.00323	-0.00536	-0.00481
^{143}Nd (60143.70c)	-0.00888	-0.00885	-0.01331	-0.0125
^{145}Nd (60145.70c)	-0.0018	-0.00166	-0.00297	-0.00255
^{147}Sm (62147.70c)	-0.00143	-0.00185	-0.00192	-0.00233
^{149}Sm (62149.70c)	-0.01558	-0.01531	-0.01649	-0.01559
^{150}Sm (62150.70c)	-0.00078	-0.0008	-0.00164	-0.00129
^{151}Sm (62151.70c)	-0.00584	-0.00431	-0.00756	-0.00514
^{152}Sm (62152.70c)	-0.00211	-0.00211	-0.0033	-0.00272
^{151}Eu (63151.70c)	-0.00027	-0.00142	-0.00023	-0.00142
^{153}Eu (63153.70c)	-0.00152	-0.00149	-0.00294	-0.00261
^{155}Gd (64155.70c)	-0.00352	-0.00681	-0.00727	-0.01304

Note: MCNP = Monte Carlo N-Particle.

[a] Monte Carlo uncertainty is less than 0.00013 Δk for FP&MA worths.

Table C.11. FP&MA worths using MCNP6 (new release) and ENDF/B–VII.1 data

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
	FP&MA worths (Δk)[a]			
All FP&MA	-0.06663	-0.0684	-0.10099	-0.10268
All MA	-0.00629	-0.00615	-0.01182	-0.01101
All FP	-0.05954	-0.06127	-0.08662	-0.08883
^{236}U (92236.80c)	-0.00391	-0.0037	-0.00576	-0.00503
^{237}Np (93237.80c)	-0.00208	-0.00237	-0.00456	-0.00468
^{243}Am (95243.80c)	-0.00022	-0.00025	-0.00117	-0.00098
^{95}Mo (42095.80c)	-0.00094	-0.00095	-0.0018	-0.00149
^{99}Tc (43099.80c)	-0.00211	-0.00214	-0.00367	-0.0033
^{101}Ru (44101.80c)	-0.00086	-0.00068	-0.00132	-0.00125
^{103}Rh (45103.80c)	-0.00645	-0.00629	-0.00976	-0.00878
^{109}Ag (47109.80c)	-0.001	-0.0007	-0.0016	-0.00135
^{133}Cs (55133.80c)	-0.00328	-0.00306	-0.00535	-0.00475
^{143}Nd (60143.80c)	-0.00887	-0.00854	-0.01338	-0.01256
^{145}Nd (60145.80c)	-0.00153	-0.00144	-0.00254	-0.00235
^{147}Sm (62147.80c)	-0.00109	-0.00177	-0.00193	-0.00228
^{149}Sm (62149.80c)	-0.01568	-0.01515	-0.01648	-0.0152
^{150}Sm (62150.80c)	-0.00085	-0.00073	-0.00152	-0.00135
^{151}Sm (62151.80c)	-0.00597	-0.00451	-0.00735	-0.00519
^{152}Sm (62152.80c)	-0.00198	-0.00195	-0.00303	-0.00279
^{151}Eu (63151.80c)	-0.00015	-0.00117	-0.00025	-0.00143
^{153}Eu (63153.80c)	-0.00158	-0.00166	-0.00305	-0.00277
^{155}Gd (64155.80c)	-0.00352	-0.00669	-0.00724	-0.01316

Note: MCNP = Monte Carlo N-Particle.
[a] Monte Carlo uncertainty is less than 0.00013 Δk for FP&MA worths.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission 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 review of burnup credit analyses supporting transport and dry storage of pressurized water reactor spent nuclear fuel (SNF) in casks. ISG-8, Rev. 3 includes, among other things, guidance for addressing validation of criticality (keff) calculations crediting the presence of a limited set of fission products and minor actinides (FP&MA). Based on previous work documented in NUREG/CR-7109, ISG-8, Rev. 3 includes a recommendation to accept use of 1.5 or 3% of the FP&MA worth, depending on the criticality code and cross-section data used, to conservatively account for the bias and bias uncertainty associated with the specified unvalidated FP&MAs. This bias is applied in addition to the bias and bias uncertainty resulting from validation of keff calculations for the major actinides in SNF. The work described in this report involves comparison of FP&MA worths calculated using SCALE and Monte Carlo N-Particle (MCNP) with Evaluated Nuclear Data Files, Part B–V, –VI, and –VII based nuclear data. The comparison supports use of the 1.5% FP&MA worth bias when either SCALE or MCNP codes are used for criticality calculations, provided the other conditions of ISG-8, Rev. 3, Recommendation 4 are met. The additional conditions include that the cask design is similar to the hypothetical generic burnup credit-32 cask model and that the credited FP&MA worth is no more than 0.1 Δkeff.

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