# Impact of Cross Section Libraries on Loading Curves for Burnup Credit Criticality Safety Analysis Models[1]

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## INTRODUCTION

The goal of this work is to assess the impact of cross section libraries on loading curves previously considered for burnup credit criticality safety models [1][2] and to revise them as necessary. Efforts are underway at Oak Ridge National Laboratory (ORNL) to update the depletion code validation approach for criticality safety analysis using burnup credit for actinide and fission product nuclides in spent nuclear fuel compositions. This approach was documented in NUREG-7108 [1]. As part of these efforts, it was essential to verify the applicability of the previous reference models and to update them as necessary for use with current computer codes [3] and cross section libraries based on recent evaluated nuclear data.

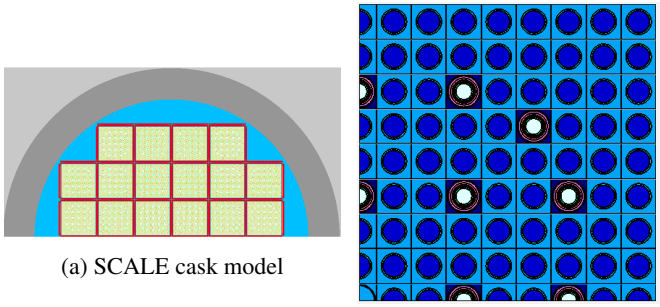
The purpose of the NUREG-7108 report was to describe an approach for establishing the depletion code bias and bias uncertainty in terms of a reactivity difference (i.e., $\Delta k_{\text{eff}}$), based on comparisons between calculated nuclide compositions and corresponding measurement data. A detailed description of the approach is given in the aforementioned NUREG report [1] and a journal article by the same authors [2]. The criticality safety reference models used and the results reported were based on SCALE 6.1.3 and the 238-group ENDF/B-VII.0 cross section library. The ongoing assessments are being performed with SCALE 6.2.4 and 252-group ENDF/B-VII.1 cross sections. Therefore, the $k_{\text{eff}}$ values of the reference models were computed using the updated SCALE version and cross section library. Based on the observed results, it was decided to update these reference models before using them to propagate the bias and uncertainties in nuclide concentrations to the $k_{\text{eff}}$ of the considered model. The investigations and findings regarding the reference models for a pressurized water reactor (PWR) spent fuel cask are summarized herein.

## MODELS DESCRIPTIONS

The criticality safety model considered is a representative PWR spent fuel cask referred to as *GBC-32* [4]. This cask contains 32 fuel assemblies with identical geometry and material specifications, and the design is a Westinghouse (W) 17×17 optimized fuel assembly (OFA). In the fuel compositions in the cask model, only nuclides of relevance to burnup credit are considered, including 12 actinide nuclides and 18 fission products. Shown in Figures 1a and 1b are illustrations of the SCALE models for the cask and the assembly, respectively. Note that for the 3D cask model, half the configuration is modeled taking advantage of the symmetry. Only 1/4 of the radial cross section of the assembly model is shown in Figure 1b. The assembly model was used for ORIGEN reactor libraries generation, where burnable absorber insertions were specified throughout irradiation period. In the cask model, burnable absorber are not included in the guide tubes. More details about the input data used with the models and a list of the included nuclides are provided in the aforementioned NUREG report [1].

The loading curve corresponding to this model was represented by seven unique pairs of enrichments and assembly-averaged burnups [1]. The burnup ranges from 5 GWd/t to 45 GWd/t, with corresponding enrichments from 2 up to 5% $^{235}$U.

The burnup-dependent nuclide concentrations (and corresponding fuel enrichments) were determined such that the cask model $k_{\text{eff}}$ value was 0.94 within the statistical uncertainty. This value allows for biases and uncertainties of 0.01 to satisfy the recommended $k_{\text{eff}}$ of 0.95 for cask criticality safety analyses [1]. A cooling time of five years was used when generating the nuclide concentrations. Also, each fuel element was discretized into 18 equal axial zones, and the burnup of each zone is computed based on the assembly average burnup and a reference burnup profile, which depends on the fuel assembly average burnup, as shown in Figure 2. Details about the burnup profile can be found in the report by Wagner et al. [5].



(a) SCALE cask model

(b) 1/4 assembly cross section

Fig. 1: Illustration of reference models.

## APPROACH

To verify the applicability of the previously established loading curve using SCALE 6.2.4 and 252-group ENDF/B-VII.1 cross sections, the following steps were followed.

1. First, ORIGEN reactor libraries were generated using a TRITON model of the assembly shown in Figure 1b. The libraries were generated for different fuel enrichments: (1, 1.5, 2, 3, 4, 5, 6, and 6.5) % $^{235}$U. Each library covers

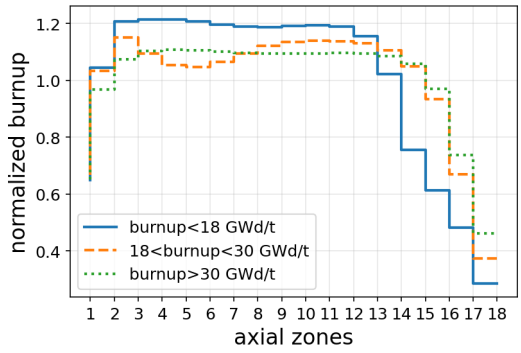



Fig. 2: Assembly axial burnup profile (axial zone one is at the bottom of the fuel element).

an average assembly burnup up to 90 GWd/t.

2. Next, the ORIGAMI tool was used to compute the nuclide concentration of each axial zone given an assembly average burnup and an initial enrichment. ORIGAMI computes the problem-dependent cross section by interpolating the ORIGEN reactor libraries discussed in the previous step. Here, the interpolation is done based on the enrichment and the axial zone burnup. The assembly average burnup was used to determine the burnup profile (see Figure 2) needed to compute the axial zone burnup. Similar to the old results, a cooling time period of five years was used for the assembly power history.

3. Finally, the resulting nuclide concentrations were applied in the cask model for each reference data point (i.e., pairs of enrichment and burnup values on the loading curve) to compute the system $k_{\text{eff}}$. The cask was modeled using the KENO-VI Monte Carlo code under the SCALE CSAS6 sequence.

Note that in the results presented in NUREG/CR-7108, the nuclide concentrations were computed using STARBUCS and ORIGEN in SCALE 6.1, and the $k_{\text{eff}}$ values were calculated using CSAS5 in SCALE 6.1. All calculations at the time were performed using 238-group ENDF/B-VII.0 data. Therefore, a slight difference was expected between the old results and the results obtained using ORIGEN, ORIGAMI, and CSAS6 in SCALE 6.2.4 and ENDF/B-VII.1 data—likely because of the change in the evaluated data from ENDF/B-VII.0 to ENDF/B-VII.1 and because of changes in the codes used between SCALE 6.1.3 and SCALE 6.2.4. To obtain insight into this expected difference, several comparisons were performed. First, the previous cask reference models that corresponded to the old loading curves were run using SCALE 6.2.4 and 238-group ENDF/B-VII.0 cross sections. Note that the fuel compositions in these models were as used in NUREG-7108. Next, the fuel compositions in the cask model were revised using SCALE 6.2.4 and 252-group ENDF/B-VII.1 libraries, using the approach described above in this section. The results of this comparison, as well as the new loading curve established, are presented in the next section.

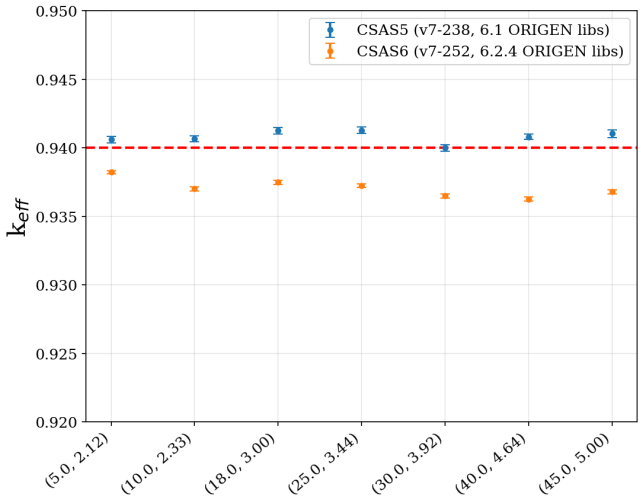



Fig. 3: $k_{\text{eff}}$ corresponding to old loading curve.

## RESULTS

Shown in Figure 3 are the $k_{\text{eff}}$ values of all reference models with the old and the new computed fuel compositions. Note that the numbers on the x-axis represent the burnup and enrichment pairs that correspond to the old loading curve. A meaningful decrease in $k_{\text{eff}}$ is observed when using the updated computer code and cross section library compared to the old models.

To understand the source of the difference in $k_{\text{eff}}$, the axial profiles of the nuclide concentrations in fuel were compared between the old and the new models. For the sake of brevity, only data for four nuclides that are the top contributors to reactivity are presented in Figures 4, 5. A good agreement is observed in case of the $^{235}$U, $^{238}$U, $^{240}$Pu. However, the difference appears to be larger in the case of $^{239}$Pu, which suggests that it is the main source of the difference between the two cases' $k_{\text{eff}}$ values. Based on the obtained $k_{\text{eff}}$ results of the

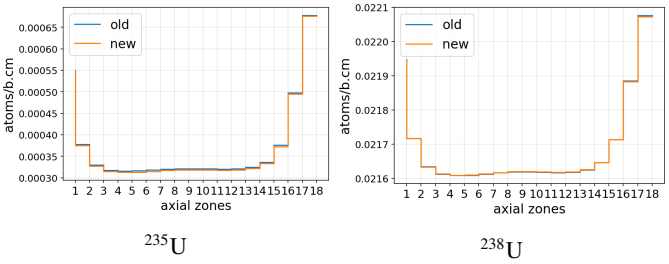
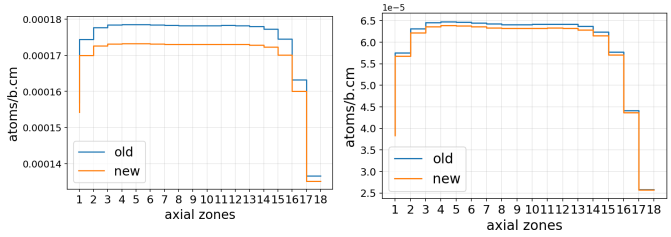



Fig. 4: Axial concentration profile for $^{235}$U and $^{238}$U.

reference models, it was decided to generate a new loading curve by updating the enrichment for each considered burnup value to better satisfy the target $k_{\text{eff}}$ value of 0.94. This update was performed by an iterative approach in which the old enrichment served as an initial guess. With this initial guess and an assumed enrichment sensitivity coefficient, new enrichments were predicted, and the cask models were rerun with nuclide concentrations corresponding to the predicted enrichments. The process is repeated with updating the sensitivity coefficient based on the new value of $k_{\text{eff}}$ until a value of 0.94




(a) $^{239}$Pu (b) $^{240}$Pu

Fig. 5: Axial concentration profile for $^{239}$Pu and $^{240}$Pu.

is reached. Moreover, the loading curve was extended to cover burnups up to 60 GWd/t. In this case, the enrichments were found using the same approach, by which the initial guess came from a linear regression using the available data points of enrichments and burnups. Note that the new enrichments exceed the 5% enrichment limit of current commercial fuel.

Shown in Figure 6 are the $k_{\text{eff}}$ values for the reference models that used the updated loading curve. Table I shows the old and new enrichment of the loading curve and the absolute difference between the enrichment values.

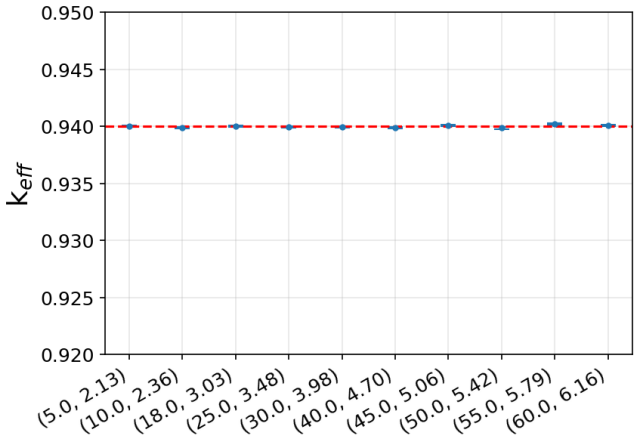



Fig. 6: $k_{\text{eff}}$ for new loading curve.

| Burnup (GWd/t) | old enrichment (wt %) | new enrichment (wt %) | difference (wt %) |
|---|---|---|---|
| 5 | 2.12 | 2.13 | 0.015 |
| 10 | 2.33 | 2.36 | 0.027 |
| 18 | 3.00 | 3.03 | 0.035 |
| 25 | 3.44 | 3.48 | 0.044 |
| 30 | 3.92 | 3.98 | 0.057 |
| 40 | 4.64 | 4.70 | 0.062 |
| 45 | 5.00 | 5.06 | 0.066 |

TABLE I: New and old loading curve enrichments comparison

## CONCLUSION

The loading curve used for criticality safety with burnup credit for a GBC-32 cask model was assessed to verify its validity with updated computer codes and nuclear data libraries. Following this assessment, a new loading curve was generated, based on the use of 252-group cross section libraries and capabilities in SCALE 6.2.4. The differences in the enrichment corresponding to a given burnup between the old and new loading curves range between 0.015 and 0.066 and increase with increasing burnup. The update of the loading curve resulted in the update of the reference cask models that are serving as a basis for the update of NUREG/CR-7108.

## ACKNOWLEDGMENT

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