

Burnup Credit Loading Curves for High-Burnup and Extended Enrichment Fuels

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INTRODUCTION

Burnup loading curves are used to determine the acceptable loading conditions and constraints for fuel assemblies in a storage configuration to show compliance with regulatory requirements regarding the subcriticality of a system [1–3].

Interest in utilizing low-enriched uranium plus (LEU+) fuel and higher-burnup fuel, such as high-assay low-enriched (HALEU), in nuclear reactors has increased. The usage of such fuel requires additional criticality safety analyses and the validation of the associated burnup credit (BUC).

The purpose of this work is to investigate the effects of the BUC analysis [1] on burnup loading curves for LEU+ and higher-burnup fuel. The work presented herein is part of a recently prepared NUREG that is currently in review [4].

MODEL DESCRIPTION

The generic BUC cask model GBC-32 with Westinghouse 17×17 optimized fuel assemblies was used in this work [5]. The cask has 32 square storage cells to fit pressurized water reactor fuel assemblies. The cells have an inner dimension of 22 cm, and stainless-steel walls and Boral panels surround them. Figure 1 shows a view of the cask and fuel.

The SCALE 6.3 code package [6] with ENDF/B-VII.1 [7] and ENDF/B-VIII.0 [8] cross-section libraries were used in this study. The SCALE TRITON “t-depl” sequence and the ORIGEN program were used for lattice physics and depletion, fuel modeling, and generation of fuel inventories for the different burnup, enrichment, and depletion conditions. The SCALE CSAS5 sequence was used for the criticality calculations to calculate the k_{eff} values and their uncertainties.

Fuel assembly average burnups up to 80 GWd/MTU and initial ^{235}U fuel enrichments from 4 to 8 wt % were considered in this study. Three sets of isotopes were studied:

- 1) Actinides only (AO): ^{234}U , ^{235}U , ^{238}U , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , ^{241}Am , and O
- 2) Actinides and 16 fission products (AFP), which include the nuclides mentioned above plus the following actinides and fission products: ^{236}U , ^{243}Am , ^{237}Np , ^{95}Mo , ^{99}Tc , ^{101}Ru , ^{103}Rh , ^{109}Ag , ^{133}Cs , ^{147}Sm , ^{149}Sm ,

^{150}Sm , ^{151}Sm , ^{152}Sm , ^{143}Nd , ^{145}Nd , ^{151}Eu , ^{153}Eu , and ^{155}Gd

- 3) Actinides and all fission products (ALL): the nuclides mentioned above plus those included in the TRITON depletion calculation with the ADDNUX=4 setting [6]. A total of 388 nuclides are included in this set.

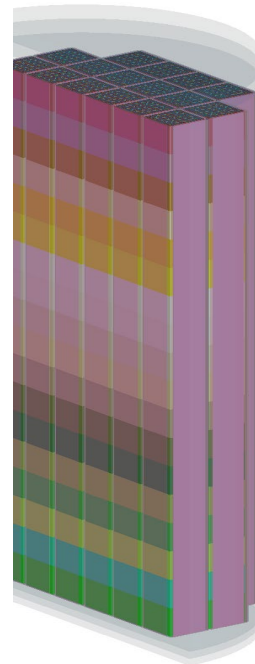


Fig. 1. GBC-32 cask model with fuel assemblies.

RESULTS

The BUC loading curves were generated for allowable fuel enrichment and burnup combinations to meet a selected target k_{eff} of 0.94 [2]. The fuel in this study was analyzed at enrichment increments of 0.5 wt % ^{235}U . Figures 2 and 3 show the BUC loading curves using the ENDF/B-VII.1 (ENDF7.1) and ENDF/B-VIII.0 (ENDF8.0) cross-section libraries, respectively. For comparison, the results of a previous study [2], which used similar depletion conditions, were added to the figures. The previous study used the ENDF/B-VII.0 cross-section libraries [9] and considered fuel enrichments up to 5 wt % ^{235}U .

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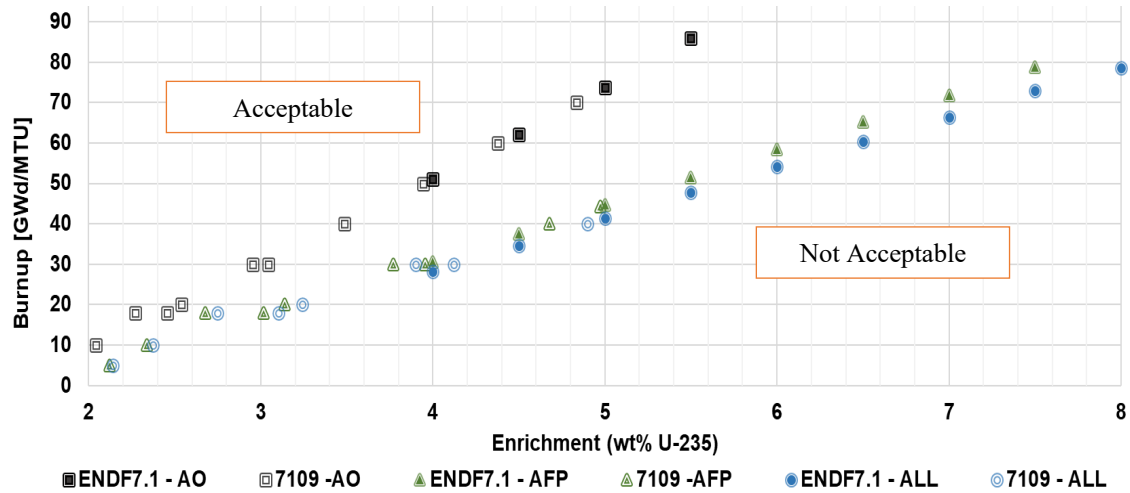


Fig. 2. Burnup loading curves using ENDF/B-VII.1 data libraries.

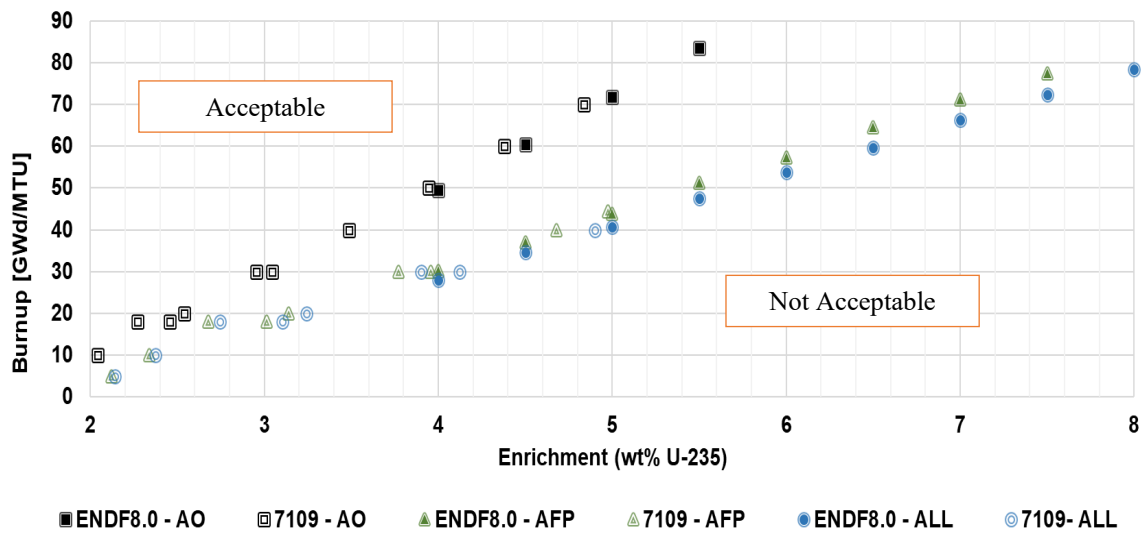


Fig. 3. Burnup loading curves using ENDF/B-VIII.0 data libraries.

The current results show a good overlap with the previous results in the 4–5 wt % ^{235}U range. As expected, the loading curves generated using the AO nuclide set are more conservative than those using the AFP and ALL. In addition, using the additional fission products in the ALL nuclide set did not result in a major difference compared with the AFP nuclide set. Acceptable fuel loading conditions are shown for 5.5 wt % ^{235}U using the AO nuclide set and for 7.5 wt % ^{235}U using the AFP nuclide set.

CONCLUSIONS

The SCALE 6.3 code package was used to generate loading curves for LEU+ fuel (up to 8 wt % ^{235}U) and higher-burnup fuel (up to 80 GWd/MTU). Both the ENDF/B-VII.1 and ENDF/B-VIII.0 cross-section libraries were used in this study. The study showed good agreement with previous studies in the 4–5 wt % ^{235}U range. For a maximum fuel burnup of 80 GWd/MTU, the acceptable loading conditions were shown for 5.5 wt % ^{235}U using the AO nuclide set and 7.5 wt % ^{235}U using the AFP nuclide set.

ACKNOWLEDGMENT

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