BURNUP CREDIT ANALYSES FOR NUCLEAR CRITICALITY SAFETY

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1. Investigation of Average and Pin-Wise Burnup Modeling of PWR Fuel, *Charlotta E. Sanders, John C. Wagner (ORNL)*

During reactor operation, fuel material composition changes because of exposure to neutron flux, among other things. These changes affect important quantities, such as the multiplication factor and the power distribution. The material changes are most commonly modeled in lattice reactor physics codes with multi-regions where each fuel pin in an assembly is modeled as one or more separate regions. In contrast, for out-of-reactor criticality safety analyses, an assembly-average fuel composition is typically used for each fuel pin (i.e., all fuel pins have the same assembly-average composition). This is done because available information for spent fuel to be loaded into a cask will be limited to the enrichment, burnup (from reactor records), and cooling time. Hence, detailed operational data will not be available, and the detail of an explicit pin-by-pin model would be overkill. Furthermore, for spent fuel storage, it is only necessary to ensure that a cask is loaded at a net reactivity that is less than that for which it was designed. The level of detail necessary to ensure subcriticality is much less than that needed to predict criticality. To evaluate the effect of this modeling approach for pressurized water reactor (PWR) fuel, a comparison of multiplication factors calculated by the 2-D lattice transport theory code HELIOS-1.6 [1] was performed using average and pin-wise model descriptions.

The PWR assemblies considered in this study are the Westinghouse (WE) 17 × 17 and the Combustion Engineering (CE) 14 × 14 fuel assemblies. The WE 17 × 17 fuel assembly was also modeled with 24 burnable poison rods (WABA) and Ag-In-Cd con-

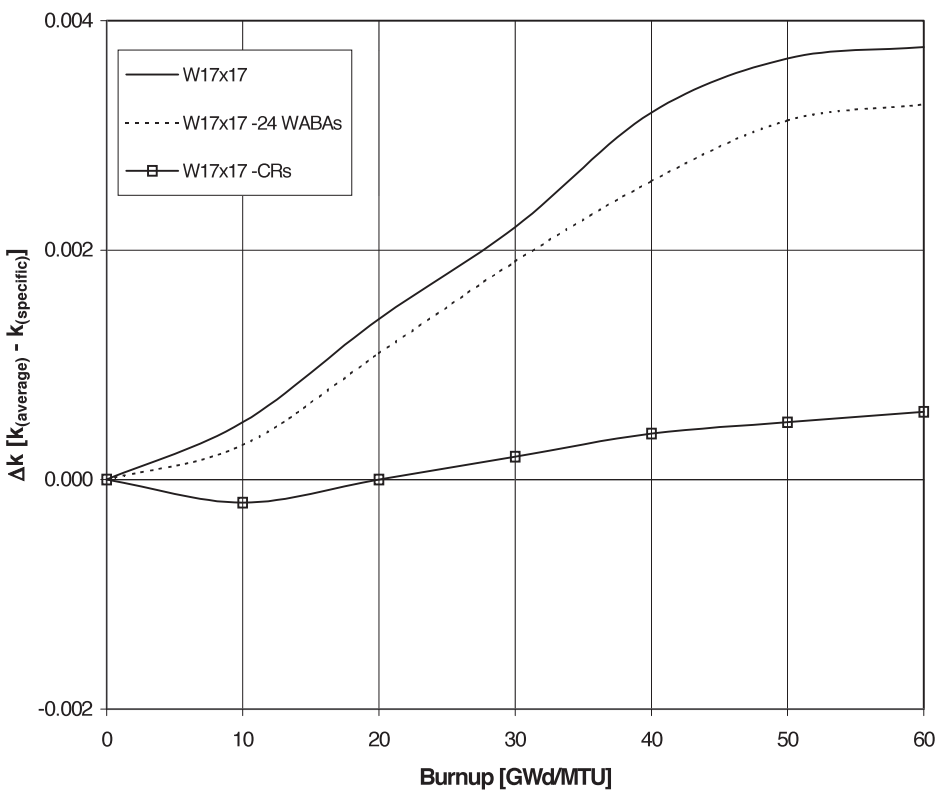

Fig. 1. Comparison of Δk values versus burnup for Westinghouse 17 × 17 fuel assemblies. The ^{235}U enrichment is 4 wt% for all cases.

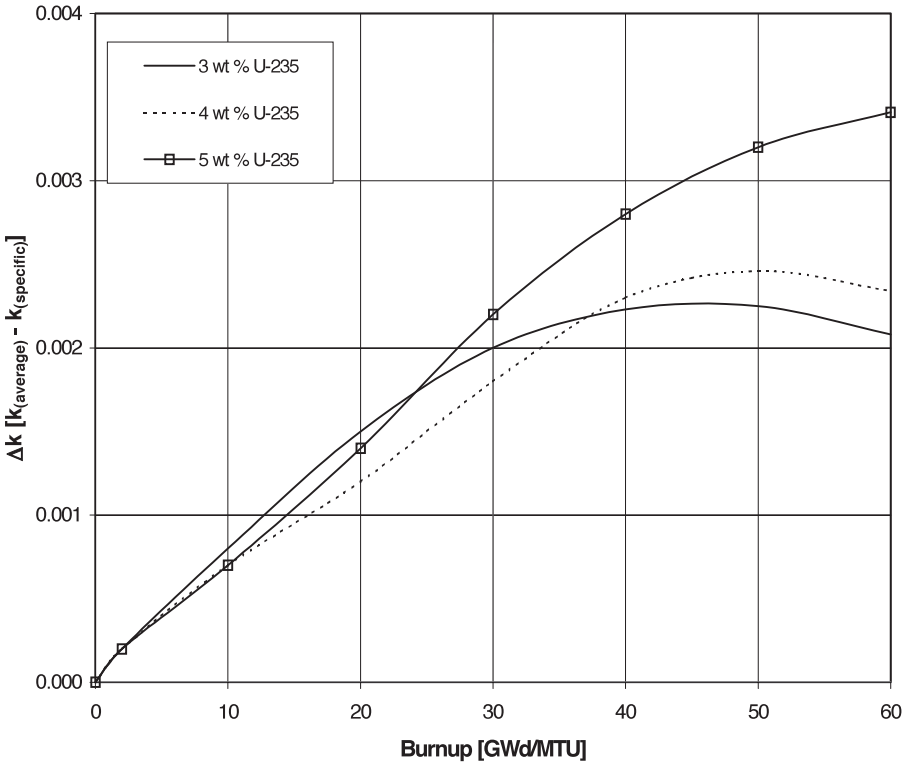

Fig. 2. Comparison of Δk values versus burnup for a CE 14 × 14 fuel assembly with varied fuel enrichment.

trol rods (CRs), with the absorber rods being present throughout the entire depletion period. A detailed description of the absorber and fuel design specifications can be found in Refs. 2 and 3. The calculations were performed for an infinite radial array of fuel assemblies and utilize all of the actinide and fission product nuclides included in the 45-group neutron cross-section library, based on ENDF/B-VI data that is distributed with the HELIOS-1.6 code package. The infinite neutron-multiplication factor, k_{inf}, was calculated as a function of burnup for out-of-reactor conditions (i.e., unborated moderator at 20°C) and zero cooling time. The depletion calculations were performed using a fuel temperature of 1000 K, moderator temperature of 600 K, a constant soluble boron concentration of 650 ppm, and a specific power of 60 MW/MTU.

When utilizing the assembly-average fuel pin modeling assumption versus the pin-wise modeling assumption, the fissile material in the lower-burned pins gets shifted to the higher-burned pins. Thus, for assembly lattices considered, fissile material is transferred from periphery fuel pins to fuel pins adjacent to guide tubes. Pins closer to guide tubes experience higher than average pin burnup due to higher moderation. The difference in the neutron multiplication factor (Δk values) between these two modeling assumptions is shown in Fig. 1 for the various Westinghouse 17 × 17 models. The results correspond to a fuel enrichment of 4 wt% ^{235}U. Note that the Δk values are lower when an absorber (e.g., WABAs and CRs) is inserted into the fuel assembly as opposed to when the fuel assembly is un-poisoned. The presence of WABAs or CRs makes the actual pin isotopic compositions more uniform, due to localized spectral hardening, and therefore the differences between the pin-wise isotopic model and average-isotopic model are less. A fuel assembly that does not contain any absorber material has varying fuel isotopic concentrations in each fuel pin, especially between inner and outer fuel pins and fuel pins near guide tubes, and consequently an assembly-average fuel composition imposed on each fuel pin is not as accurate as a pin-wise fuel composition. In each case, the use of assembly-average composition yielded higher k_{inf} values.

The Δk values between the two modeling assumptions for the CE 14 × 14 model at 3, 4, and 5 wt% ^{235}U initial fuel enrichments are shown in Fig. 2. It can be seen that the Δk values are increasing with increasing fuel enrichment. These differences are attributed to the shift of fissile material from the lower burned pin regions to the higher burned pin regions. As there is more variation in the isotopic concentrations in each fuel pin for the 5 wt% ^{235}U enrichment case than the 3 wt% ^{235}U enrichment case (due to higher fissile content in the fuel material for the 5 wt% ^{235}U enrichment case), the differences generally increase with initial enrichment.

In addition to the calculations described above, analyses were done utilizing an infinite radial array of assemblies in a poisoned storage cell, which was based on the generic 32 PWR-assembly burnup credit (GBC-32) cask, in order to study its impact on the two modeling assumptions. A physical description of the cask is provided in Ref. 4. The results showed that the Δk values increased slightly in comparison to the infinite radial array calculations for un-poisoned fuel assemblies. This was expected since the assembly-average isotopic modeling assumption effectively moves fissile material inward away from the assembly periphery, which is near the storage cell poison panels, toward the assembly center.

It can be concluded that the magnitude of the effect of composition modeling on PWR fuel, using pin-wise and assembly-average modeling descriptions, has a relatively small impact on the multiplication factor. In all cases considered, the assembly-average composition modeling resulted in larger k_{inf} values (conservative). It was also noticed that the Δk values between the two modeling assumptions were found to be increasing with increasing fuel enrichment. Further, when absorber rods were present, the Δk values were reduced.

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2. J. C. WAGNER, C. V. PARKS, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," ORNL/TM-2000/373, Oak Ridge National Lab. (2002).

3. C. E. SANDERS, J. C. WAGNER, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit," ORNL/TM-2001/69, Oak Ridge National Lab. (2002).

4. J. C. WAGNER, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit," NUREG-6747 (ORNL/TM-2000/306), U.S. Nuclear Regulatory Commission/Oak Ridge National Lab. (Oct. 2001).

2. Computational Benchmark of SAS2D Against Spent Fuel Samples from the Takahama-3 Reactor, *Charlotta E. Sanders, Mark D. DeHart (ORNL)*

SUMMARY

The accurate prediction of nuclide compositions in spent nuclear fuel is necessary in order to evaluate important quantities, such as the neutron multiplication factor and radionuclide concentrations and toxicities for assessment of long-term environmental waste management concepts. The Oak Ridge National Laboratory (ORNL) has recently completed development of a two-dimensional (2-D) depletion sequence, SAS2D. Benchmark simulations have been performed with SAS2D against spent fuel radiochemical assay measurements from the Kansai Electric Ltd. Takahama-3 reactor.

The 2-D depletion sequence SAS2D [1] is a control module within the SCALE code system. SAS2D uses the 2-D arbitrary geometry, S_n theory code NEWT [2] to provide 2-D fluxes for a user-specified configuration. Depletion calculations are then performed for as many different materials as desired within the assembly configuration using multiple ORIGEN-S [3] calculations. SAS2D is currently not publicly available but will be released in SCALE 5 later this year.

Takahama-3 is a pressurized-water reactor (PWR) with 17 × 17 fuel lattice assemblies. Spent fuel samples were obtained from two assemblies, NTG23 and NTG24, irradiated for 2 and 3 cycles, respectively. Both fuel assemblies contain 14 integral burnable gadolinia-bearing (Gd_2O_3) fuel rods containing 2.6 wt% ^{235}U and 6.0 wt% Gd_2O_3 while the standard fuel rods contain 4.11 wt% ^{235}U enrichment. Spent fuel samples were obtained from two standard fuel rods (SF95 and SF97) and one gadolinia-bearing fuel rod (SF96). A total of five samples were each taken from various axial locations of fuel rods SF95 and SF96, and six samples were taken from fuel rod SF97. Only one sample from each fuel rod was selected for this study. The SF95-3 fuel sample has a burnup of 35.52 GWd/MTU and was measured at an axial location of 88.1 cm (the distance is measured from the top of the active region of the fuel rod). The SF95-3 sample is taken from a fuel pin located in the corner of the fuel assembly, which had

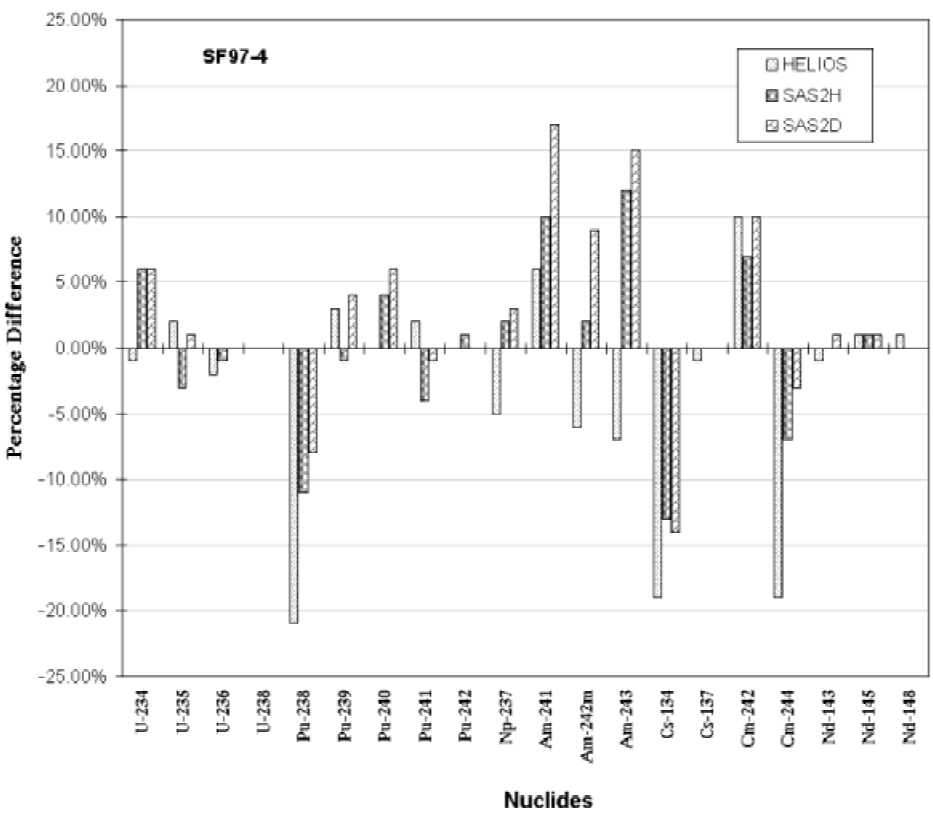

Fig. 1. Percentage difference between calculated and measured nuclide concentrations for the SF97-4 fuel sample. The calculations were performed with the HELIOS, SAS2H, and SAS2D code systems.