BURNUP CREDIT ANALYSES FOR NUCLEAR CRITICALITY SAFETY

Sponsored by the Nuclear Criticality Safety Division

Session Organizer: Mark DeHart (ORNL)

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-

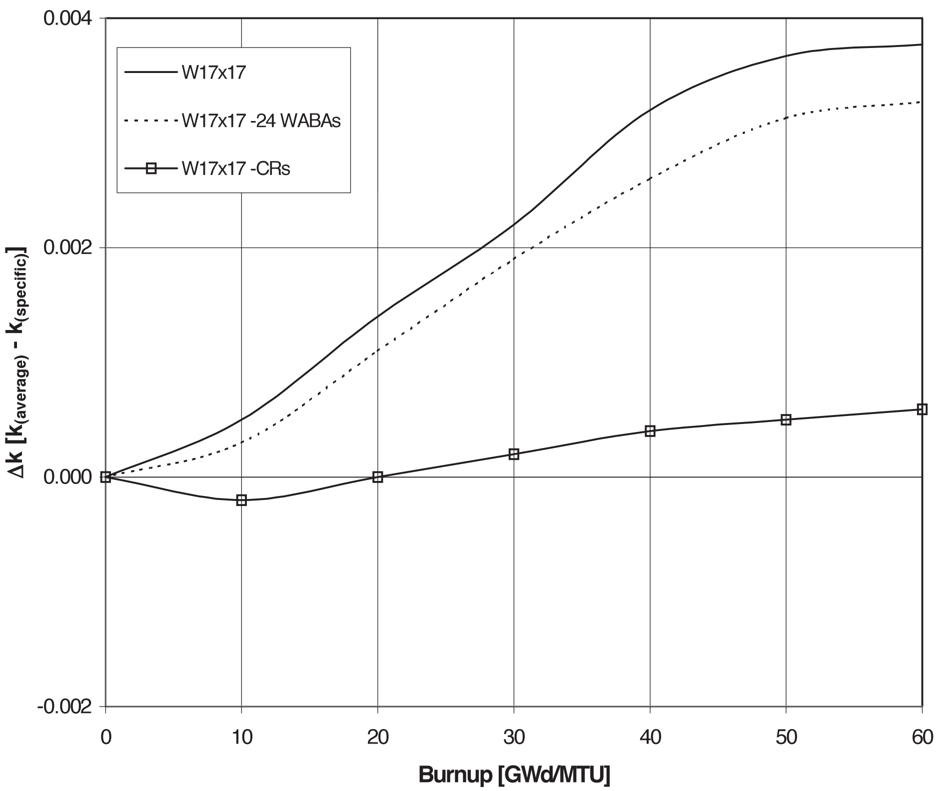

Fig. 1. Comparison of Δk values versus burnup for Westinghouse 17×17 fuel assemblies. The ^{235}U enrichment is 4 wt% for all cases.

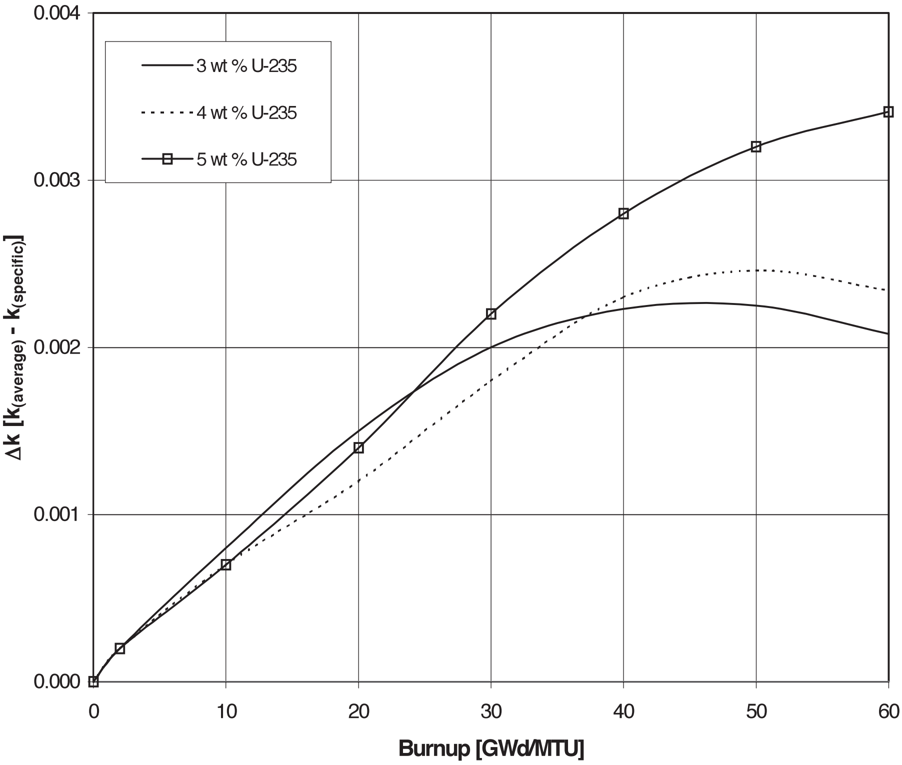

Fig. 2. Comparison of Δk values versus burnup for a CE 14 × 14 fuel assembly with varied fuel enrichment.

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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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

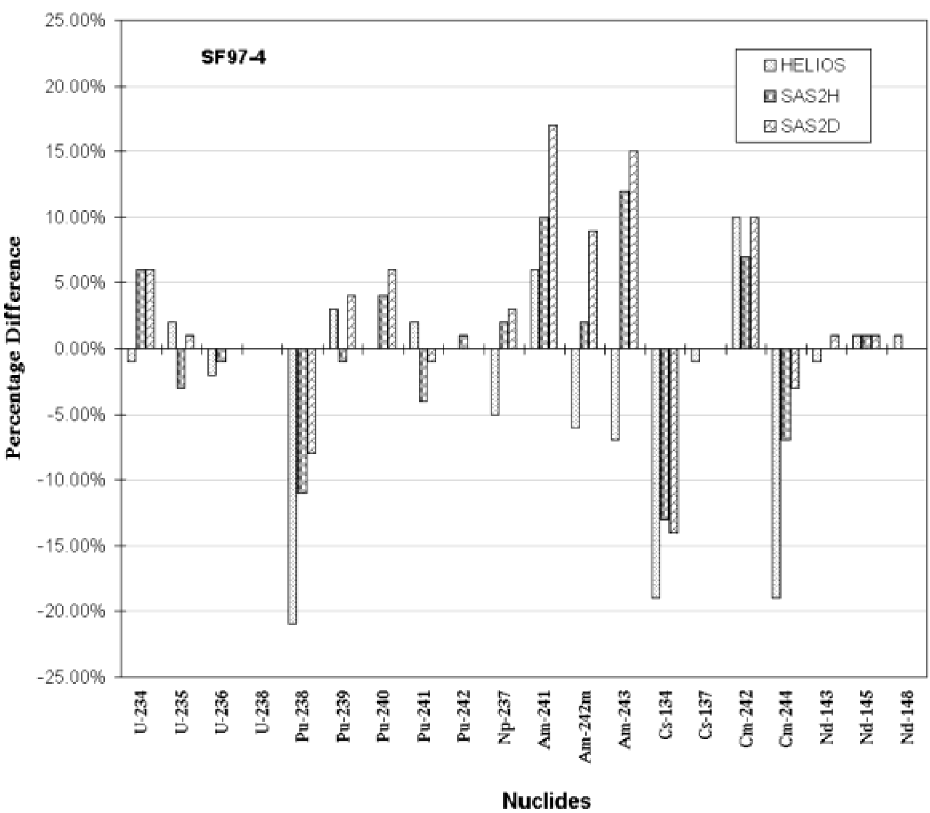

Fig. 1. Percentage difference between calculated and measured nuclide concentrations for the SF97-4 fuel sample. The calculations were performed with the HELIOS, SAS2H, and SAS2D code systems.

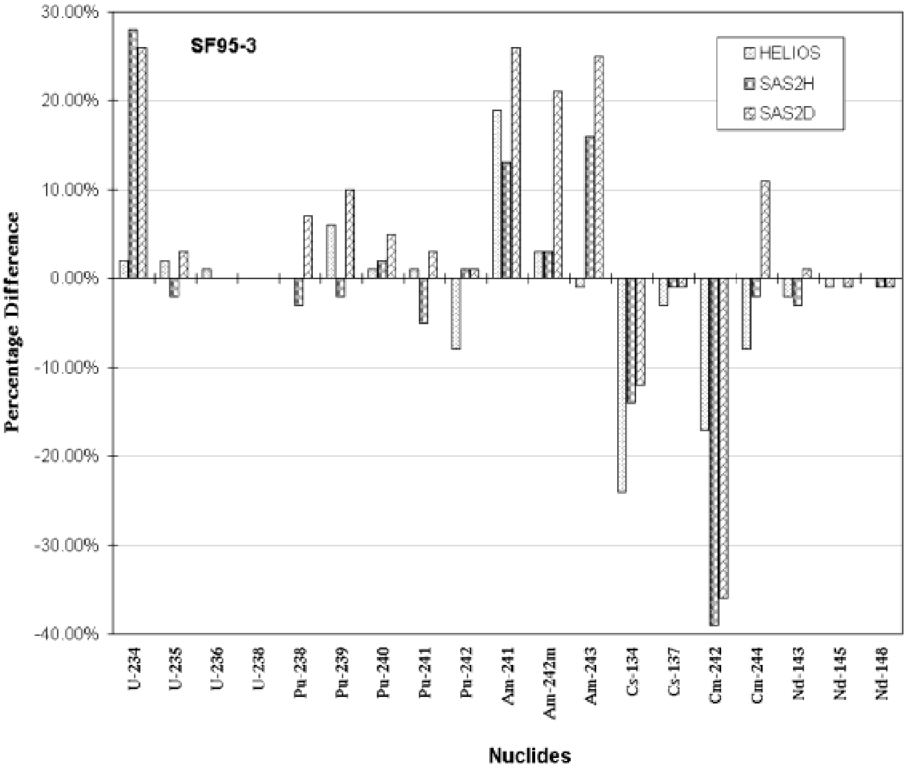

Fig. 2. Percentage difference between calculated and measured nuclide concentrations for the SF95-3 fuel sample. The calculations were performed with the HELIOS, SAS2H, and SAS2D code systems.

not been subject to local flux perturbations from the water holes or the burnable poison rods in the assembly during irradiation (i.e., all adjacent rods were standard fuel pins). The SF96-3 fuel sample has a burnup of 28.20 GWd/MTU and was extracted from an axial location of 85.6 cm. Note that the SF96-3 fuel sample was obtained from a gadolinia-bearing rod. The SF97-4 fuel sample has a burnup of 47.03 GWd/MTU and was located at an axial location of 183.9 cm, near the mid-plane of the rod. The SF97 fuel rod was a standard fuel rod, similar to SF95, located at the outer edge of the assembly. A detailed description of the Takahama-3 reactor and fuel design specifications, as well as a summary of several basic parameters of the three measured spent fuel rods, can be found in Ref. 4.

Comparison of measurements and calculations of sample SF97-4 for various isotopes are presented in Fig. 1 in terms of percentage difference between measured and calculated isotopic concentrations. The comparisons show that overall the SAS2D predictions agree very well with the measurements. Figure 1 also displays the results of previously performed calculations with the SAS2H and HELIOS code systems [4] for the same fuel sample. It is interesting to note that the SAS2D predictions, for the most part, are more accurate than both SAS2H and HELIOS, especially for U and Pu isotopes. There are some disagreements, however, between the calculated-to-experimental ratios for the Am and Cm nuclides. Since this is the case for all three codes, these differences could be due to data rather than modeling discrepancies. Similar trends were observed for the SF95-3 sample displayed in Fig. 2. Figure 3 displays the percentage difference between measured and calculated isotopic concentrations for sample SF96-3 (Gd-rod). While SAS2D shows better agreement than the SAS2H and HELIOS codes, there are again some large differences between the calculated-to-experimental ratios for the Am and Cm isotopes.

The radiochemical isotopic assay data for the Takahama-3 reactor have been applied to validate the isotopic predictions using SAS2D. The SAS2D predictions for the standard fuel samples SF95 and SF97 were in good agreement with the measurements. Further, the SAS2D predictions were on the similar level of accuracy as those of the SAS2H and HELIOS code systems. In fact, SAS2D proved to be more accurate in several cases, especially for most of the U and Pu isotopes. Future improvements of SAS2D include the usage of the CENTRM code [5], in lieu of the NITAWL sequence in the SCALE cross-section processing scheme, in order to provide resolution of spatially varying cross sections. This way spatially varying media (e.g., absorber rods) can be modeled more accurately, which should also reduce the differences noted with the SF96 Gd-rod sample.

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3. O. W. HERMANN, R. M. WESTFALL, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Vol. II, Sec. F7 in "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Rev. 6, ORNL/NUREG/CSD-2/R6, Vols. I, II, and III, Oak Ridge National Laboratory (May 2000) (Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545).

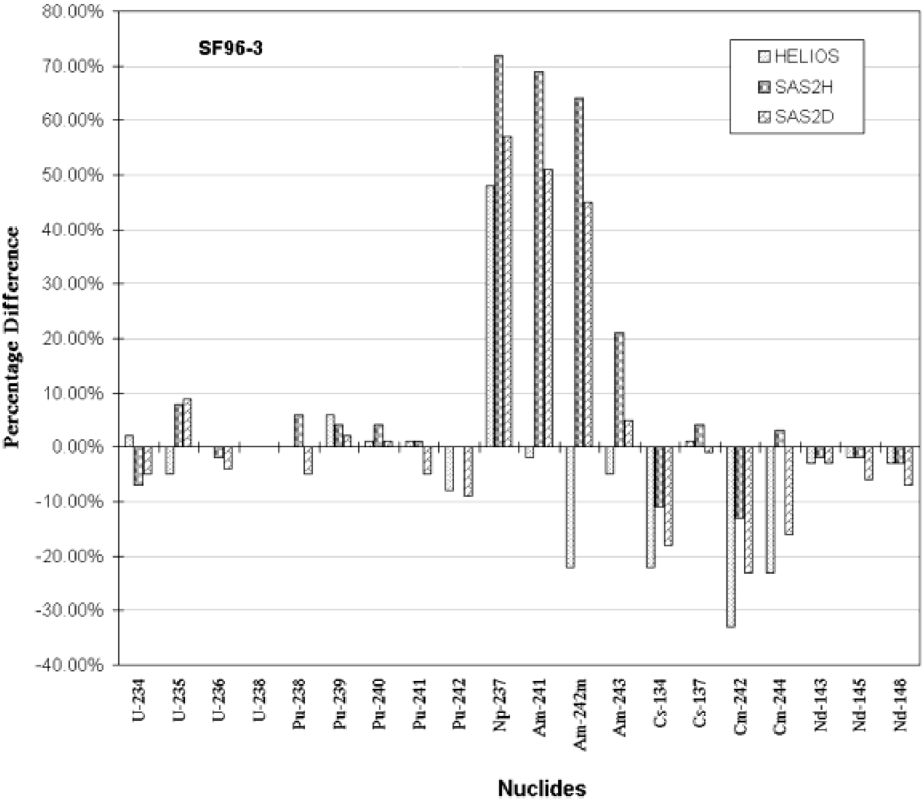

Fig. 3. Percentage difference between calculated and measured nuclide concentrations for the SF96-3 fuel sample. The calculations were performed with the HELIOS, SAS2H, and SAS2D code systems. Note that the SF96-3 fuel sample is taken from a Gd-rod.

4. C. E. SANDERS, I. C. GAULD, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor," ORNL/TM-2001/259, Oak Ridge National Laboratory (2002).

5. M. WILLIAMS, M. ASGARI, "Computation of Continuous-Energy Neutron Spectra with Discrete Ordinates Transport Theory," *Nucl. Sci. Eng.*, **121**, 173 (1995).

3. PWR Burnup Credit Using Both Belts and Suspenders, *Dale Lancaster (NuclearConsultants.com)*

INTRODUCTION

In criticality licensing of spent nuclear fuel containment systems, taking credit for the decrease in fuel reactivity due to in-core burnup is known as burnup credit. Burnup credit has been successfully used in spent fuel pool criticality licensing and has been used as a loading criterion in dry fuel storage in the United States. It has also been used in Europe. Burnup credit has not yet been licensed for transport of commercial spent nuclear fuel in the United States. There has been much work performed to allow application for burnup credit for transport [1, 2], but no license applications have been submitted by the time of this writing (February 2002).

The current position of the U.S. Nuclear Regulatory Commission is given in ISG8 Rev.1 [2]. This position gives credit only for the change in actinide concentration and requires conservatism in a number of independent steps. ISG8 Rev.1 rests heavily on the limited clean experimental data of critical experiments using MOX and fresh UO_2 fuel and on chemical assays of spent nuclear fuel to justify the isotopic content of fuel as a function of burnup. It is well accepted that the "actinide-only" approach yields much less credit than the real change of reactivity, but proof of the magnitude of the conservatism is not possible without further experiments.

Unfortunately, "actinide-only" burnup credit appears to be insufficient for fuel that is being placed in the cores today. Figure 1 shows a loading curve for a typical 32 assembly transport cask. Although the loading curve is acceptable for much of the older fuel with enrichments of about 3.2 wt% U-235, it is clear that it does not cover fuel with initial enrichments greater than 3.6 wt%. Refinements in the method may allow loadings of up to 4.0 wt%, but then significant changes in methods would have to be considered. The current solution to the loading curve problem is to add absorber rod inserts into the fuel assemblies. The number of assemblies that would need these inserts is fairly limited, 4 to 16 of the 32 assemblies in the cask. These add cost and decrease the number of available positions for disposal of the utilities spent fuel inserts such as burnable absorbers. An alternative solution that has been investigated is to derate the capacity of the cask from 32 assemblies to 30, 28, or 24 assemblies [3].

This summary introduces a new approach to burnup credit for transport that utilizes the dry nature of the spent fuel casks (the belt) but also requires adequate burnup in case of unexpected conditions (the suspenders).

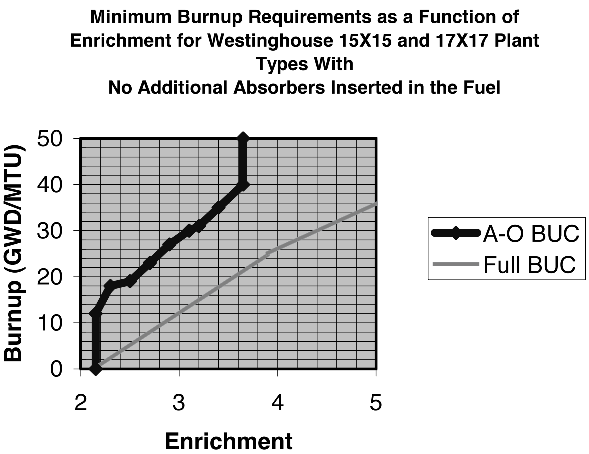

Fig. 1. Loading curve for a typical 32 assembly transport cask using Actinide-Only Burnup Credit (A-O BUC) or Moderator Exclusion with Burnup Credit (Full BUC).

THE LAW

The law in 10CFR71.55b requires that the criticality analysis for a transport cask assume that the cask is flooded with pure water. However, 10CFR71.55c states the following:

> The Commission may approve exceptions to the requires of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.

Many of the cask designs now have double lids so it should be possible to meet the "special design feature" requirement of the law. The cask systems also generally have two independent containers (the canister and overpack) or a thick walled cask body.

Leak tests are already generally performed prior to shipping.

The casks filled with commercial fuel but no water are very subcritical. One could argue that no burnup credit is needed. However, it is always nice to have suspenders to support the criticality safety positions.

BURNUP CREDIT WITH MODERATOR EXCLUSION

Currently, PWR spent fuel pools are subcritical due to the dissolved boron in the pool. They are however required to show that if there were no dissolved boron there would still not be a criticality [4]. The burnup credit proposed here uses the same type of argument. It will be required to show that if the cask were flooded with pure water, it would not be critical. Current spent fuel pool analysis allows for the criticality criteria a k_{eff} of 1.0. This is often referred to as boron credit of about 500 ppm. For transport burnup credit using moderator exclusion, this proposal assumes the criticality criterion is k_{eff} and must be less than 0.95.

The burnup credit analysis for transport burnup credit should follow the methods currently used for spent fuel pools [4]. This method is known as full burnup credit and uses all the isotopes in the spent nuclear fuel.

Figure 1 shows the results of this method applied to the same typical 32 assembly cask. As can be seen in Fig. 1, fuel with a reasonable discharge burnup can be placed in the cask without addition of absorber inserts or derating the cask.

DISCUSSION AND CONCLUSION

In order for the belt (moderator exclusion) and suspenders (burnup credit) to be applicable the cask cannot be loaded or unloaded in an unborated pool. This clearly eliminates BWRs. Currently, the Yucca Mountain design unloads the fuel dry.

Moderator exclusion with burnup credit does not utilize burnup credit as its main method of being subcritical. Therefore, the fuel does not require a burnup verification measurement prior to shipment as required by the NRC and IAEA. This is consistent with current use of burnup credit in spent fuel pools.

From a risk informed licensing point of view, moderator exclusion with burnup credit has a small but positive benefit. Moderator exclusion with burnup credit requires less fuel handling than current burnup credit approaches since no burnup measurement is needed and no additional inserts are needed. Less fuel handling is a small safety benefit along with a small reduction in operational radiation exposure. The risk of flooding is extremely low. The recent study of the risks associated with transport [5] shows that the risk of train accident resulting in the cask landing in significant water to be about $2*10^{-6}$ per shipment. Further, it is clear that impact with water would not be a reasonable mechanism to fail the cask. Also remember that if the cask is flooded, it will not go critical but have a best estimate k_{eff} of less than 0.95.

In conclusion, moderator exclusion with burnup credit is a safe approach for licensing transport casks. Criticality is not credible with this approach of belts and suspenders.

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5. J. L. SPRUNG et al., "Reexamination of Spent Fuel Shipment Risk Estimates," NUREG/CR-6672 (SAND2000-0234), Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission (Mar. 2000).