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ABSTRACT
In 1999 the U.S. Nuclear Regulatory Commission (U.S. NRC) initiated a research program to support the development of technical bases and guidance that would facilitate the implementation of burnup credit into licensing activities for transport and dry cask storage. This paper reviews the following major areas of investigation: (1) specification of axial burnup profiles, (2) assumption on cooling time, (3) allowance for assemblies with fixed and removable neutron absorbers, (4) the need for a burnup margin for fuel with initial enrichments over 4 wt %, and (5) evaluation of assay data and critical experiments. The capabilities of a new computational tool that facilitates the performance and coupling of the depletion and criticality analyses needed for burnup credit are also discussed.

I. INTRODUCTION

The concept of taking credit for the reduction in reactivity due to irradiation of nuclear fuel (i.e., fuel burnup) is commonly referred to as burnup credit. The reduction in reactivity that occurs with fuel burnup is due to the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides (non-fissile actinides and fission products). Historically, criticality safety evaluations for transportation packages have assumed the fuel contents to be unirradiated fuel compositions. In July 1999 the U.S. NRC Spent Fuel Project Office (SFPO) issued Revision 1 of Interim Staff Guidance 8 (ISG8) to provide staff recommendations for the use of burnup credit for storage and transport of pressurized water reactor (PWR) spent fuel.[1] Subsequently, the recommendations of ISG8 were included in the staff Standard Review Plan for transportation casks.[2]

Since the issuance of ISG8 in July 1999, the U.S. NRC Office of Regulatory Research (RES) has sponsored Oak Ridge National Laboratory (ORNL) to help develop expanded guidance relative to selected elements of ISG8, to develop a technical basis for staff consideration of potential revisions of ISG8, and to implement software enhancements that can facilitate the use of computational methods in safety analyses. A baseline report[3] was prepared to review the status of burnup credit and to provide a strawman prioritization for areas where additional guidance, information, and/or improved understanding were considered to be beneficial to the effective implementation of burnup credit in transport and dry storage casks. As a result of the initial review and input from industry and licensing staff, the focus areas for the NRC research program were established and will be discussed below.

II. AXIAL BURNUP PROFILE

As indicated by ISG8, the axial burnup profile in a spent fuel assembly is an important component of the safety analysis. However, ISG8 provides little information on an acceptable approach to address this issue in the licensing application. Thus, the research program has sought to develop and propose initial guidance that can be readily implemented by industry and readily reviewed by NRC staff. To this end ORNL staff has reviewed and evaluated[4] a database of 3169 axial burnup profiles from ~1700 different assemblies. The database[5] was developed using information from 20 different U.S. PWRs representing 106 cycles of operation through the mid-1990s. Although the database represents only 4% of the assemblies discharged through 1994, the ORNL review indicates the database provides a good statistical representation of discharged assemblies in terms of fuel vendor/reactor design, types of operation (i.e., first cycles, out-in fuel management and low-leakage fuel management), burnup and enrichment ranges, and use of burnable absorbers. For burnup and enrichment values beyond the current limits of ISG8 (40 GWd/MTU and 4.0 wt %), expansion of the existing database would be desirable to increase the number of profiles representing that regime. However, Ref. 4 indicates that the bounding profile from intermediate burnup ranges do bound the available profiles at higher burnups. Consequently, the existing database may be adequate for burnups beyond 40 GWd/MTU; additional work is needed to better understand the phenomena.

Previous work[6] identified the axial profiles within the database that provide the highest neutron multiplication factors (k_{eff}) over selected burnup ranges. This information was used to propose artificial bounding profiles for each burnup range. Figure 1 shows the spread of k_{eff} values that result from the set of profiles available from a selected burnup range, together with the actual bounding profile from the database and the proposed (artificial) bounding profile from Ref. 6. The figure shows the mean k_{eff} value and indicators for 1, 2, and 3 standard deviations. An examination of the calculated k_{eff} values reveals that, for each of the 12 burnup ranges, the k_{eff} value associated with the actual bounding axial profile is more than 3 standard deviations above the mean and, in most cases, is more than 5 standard deviations above the mean. In other words, the limiting profiles can be considered statistical outliers, as opposed to being representative of typical spent nuclear fuel (SNF) profiles. Consequently, one can infer that there is a very small probability for the existence of other profiles that are notably more reactive than the limiting profile (determined from the database). When one considers that the limiting profiles are based on statistical outliers and that these limiting profiles will be applied to all assemblies in a burnup credit cask, it is clear that this approach should provide conservative results in comparison to realistic loadings of a SNF cask. Thus this publicly available database is believed to be an appropriate source for selecting bounding axial burnup profiles to be used in a safety analysis.

III. COOLING TIME

ISG8 recommends that safety analyses be performed at a fixed cooling time of 5 years. Figure 2 shows the trend of k_{eff} for a 32-element, generic burnup credit cask design[7] (GBC-32). For burnup-credit criticality safety analyses performed at 5 years, increased cooling times result in an increasing conservative safety margin out to ~50 years. The additional benefit for cooling times between 50 and 100 years is insignificant. A cooling time of 40 years provides a k_{eff} value that approximately equates to the k_{eff} value at 200-year cooling, which might be considered a practical lifetime for dry storage and transport casks. Thus this rationale leads to a conclusion that cooling times up to 40 years can be assumed in developing the safety basis. To address concerns with use of storage casks beyond the assumed 200-year storage time and to lay a consistent foundation that enables future extension beyond the ISG8 actinide-only recommendation, it has been suggested that a value of 10 years be assumed as the cooling time limit for safety analysis. The rationale is that the best-estimate results (for k_{eff} at a 10-year cooling time) are always greater than the maximum k_{eff} in the secondary peak (10,000-to-30,000-year time frame).

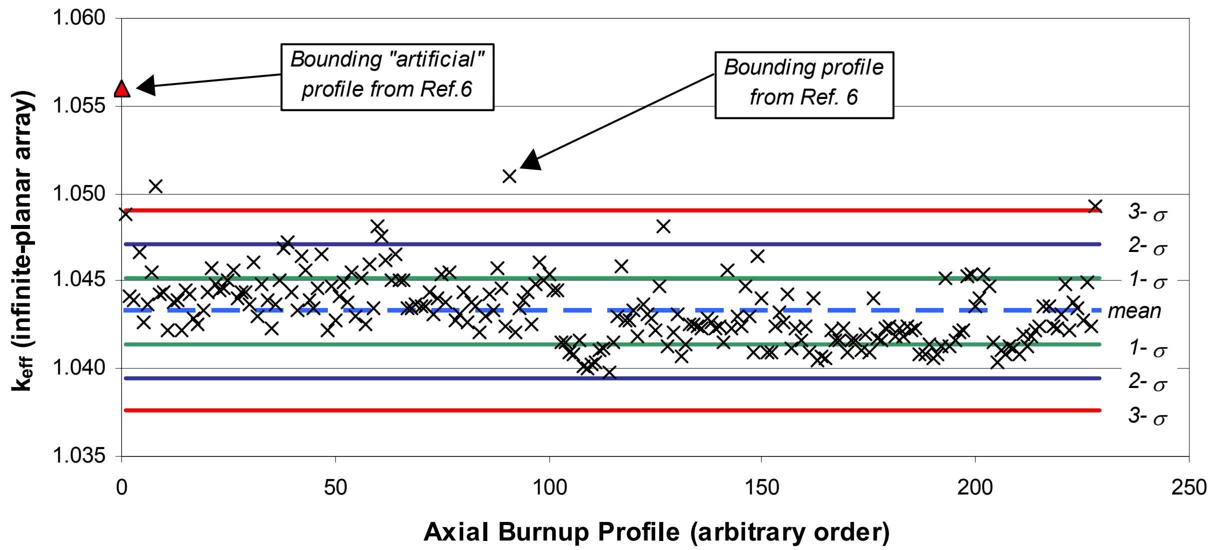

Figure 1. Values of k_{eff} for an infinite planar array as a function of database axial profiles for 38–42 GWd/MTU.

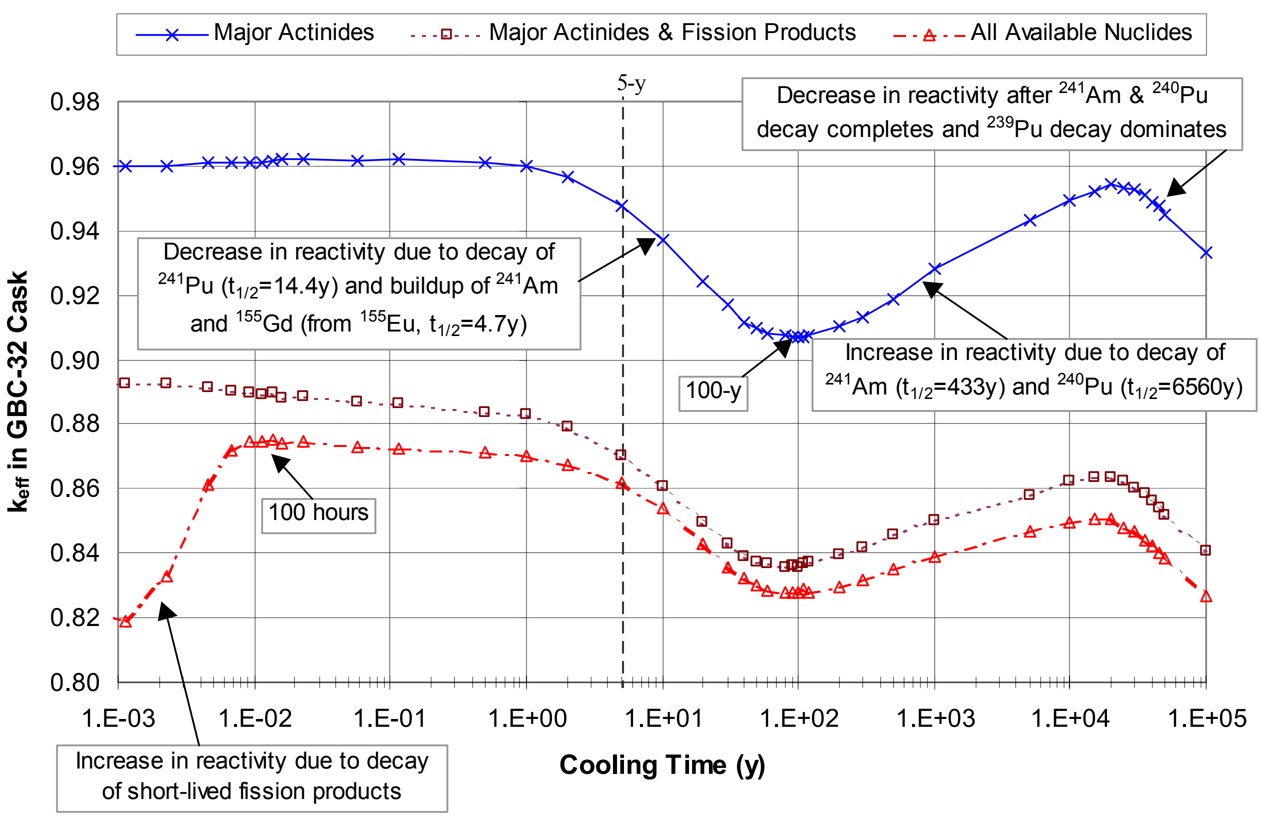

Figure 2. Values of k_{eff} in the GBC-32 cask as a function of cooling time for 4.0-wt % fuel burned to 40 GWd/MTU.

IV. FIXED AND REMOVABLE ABSORBERS

Assemblies exposed to fixed neutron absorbers [integral burnable absorbers (IBAs)] and removable neutron absorbers [burnable poison rods (BPRs) and control rods (CRs)] can have higher k_{eff} values than assemblies which are not exposed because the presence of the absorber will harden the spectrum and lead to increased ^{239}Pu production and reduced ^{235}U depletion. Since this effect had not been fully quantified at the time ISG8 was issued, the NRC recommendation in ISG8 was to restrict the use of burnup credit to assemblies that have not contained IBAs or BPRs during any part of their exposure. Concern was also indicated regarding the effect of CRs. The restriction on burnable absorbers (IBAs and BPRs) eliminates a large portion of the spent fuel from being loaded in a burnup credit cask. To provide a technical basis for potential change to this guidance, ORNL has performed investigations[8–10] that quantify how the k_{eff} of a discharged assembly would change due to exposure to BPRs, IBAs, and CRs. A comprehensive range of assembly designs, absorber loadings, and exposure history (for BPRs and CRs) was used to determine the impact on the k_{eff} of spent fuel. The studies show that exposure to BPRs can cause the k_{eff} to increase a maximum of 3% when the maximum number of BPRs and/or the maximum absorber loading is assumed for the maximum exposure time. More typical absorber loadings and exposures lead to increases of <1% Δk. By comparison, except for one IBA type where the increase was a maximum of 0.5% Δk, the IBAs actually provide a decrease in k_{eff} relative to assemblies not exposed to IBAs. References 8–9 provide a base characterization for the effect of burnable absorbers on spent fuel and indicate that a depletion analysis with bounding BPR loadings and exposure limits should provide an adequate bounding safety basis for fuel with or without burnable absorbers.

For the parametric study to quantify the effect of CR exposure, the results of Ref. 10 show that even for significant burnup exposures (up to 45 GWd/MTU), minor axial CR insertions (e.g., <20 cm) result in an insignificant effect (less than 0.2% Δk) on the k_{eff} of a burnup credit cask. Consequently, since U.S. PWRs do not use CRs to such a significant extent, use of a bounding BPR depletion model will also bound the potential effect of CRs on discharge reactivity.

V. LOADING OFFSET FOR HIGH INITIAL ENRICHMENTS

Currently, ISG8 limits credit for burnup to 40 GWd/MTU and initial enrichments to 4 wt %, although allowance for initial enrichments up to 5 wt % is permitted with an added burnup margin applied at loading. The major reason for these recommended limitations is the lack of chemical assay data for higher burnups and enrichments. The present experimental database of public domain actinide assay data consists largely of samples from older fuel assembly designs with enrichments below 3.5 wt %, and contains only one measurement for fuel above 3.4 wt % (a 3.89 wt % sample with a low burnup of 12 GWd/MTU). Only seven of the approximately 50 samples had BPRs present during irradiation. The loading offset of ISG8 provides a means of extending the usefulness of ISG8 to include spent fuel with initial enrichments above 4 wt % using an engineering approach to compensate for potentially larger uncertainties. Extending the ISG8 recommendations beyond the current limits would require additional experimental data and/or work to extrapolate the code bias and uncertainties obtained from comparison to the current measured assay data. Efforts to justify extrapolation of the bias and uncertainties have proven to be challenging because of the limited amount of experimental data and the large number of different parameters that can affect the bias. Several studies[11,12] do suggest, however, that the effect of enrichment on isotopic uncertainties is minimal. Published French results[11] for Gravelines spent fuel using French computational methods and JEF cross-section data indicate a level of agreement that is comparable to that of lower-enrichment fuel. In addition, sensitivity-based methods have been applied at ORNL to assess the influence of nuclear data bias and uncertainties on the isotopic compositions and the k_{eff} of a spent fuel storage cask.[12] These studies indicate that there is a strong correlation between spent fuel systems with a constant enrichment-to-burnup ratio. The results suggest that existing isotopic assay data may be highly applicable to regimes well beyond that of the data and that the basic depletion phenomena do not change significantly with relatively minor increases in enrichment (i.e., from 4 to 5 wt %).

Indeed, a recent study at ORNL using new isotopic assay data from the Takahama 3 reactor[13] supports these observations. The Takahama measurements include assay data for 4.1 wt % fuel with a burnup up to 47 GWd/MTU and includes an extensive number of burnup credit actinides and fission products. The results indicate there is no significant increase in the uncertainty of the neutron multiplication factor due to predicted isotopic uncertainties for higher enrichment or burnup fuel. Work is progressing to combine the limited quantity of new assay data with the existing assay database and the ORNL sensitivity-based methods to provide additional evidence to support predictions beyond the range where the majority of experimental data exist.

VI. EVALUATION OF EXPERIMENTAL DATA

A review and evaluation of existing and proposed experimental data is underway at ORNL. The purpose is to rank the relevance of experiments for methods validation using quantitative criteria and help identify experimental needs. Existing (albeit some are proprietary) experimental data include chemical assays of spent fuel nuclide inventories, critical experiments performed with fresh fuel in cask-like geometries, reactivity-worth measurements, subcritical experiments, and critical configurations in operating reactors. The potential value and limitations of each of these types of experiments were reviewed in Ref. 3. To assist in understanding and assessing the value of these experimental types, sensitivity/uncertainty (S/U) methods discussed in Ref. 14 are being used to provide information on the strengths and potential limitations of various types of experiments relative to validation needs for burnup credit. Existing fresh fuel (UO_2 and mixed-oxide) critical experiments, reactor-critical configurations, reactivity-worth experiments, and measured chemical assay data have been studied with prototypic S/U methods.[15] Recommendations are being developed on the experiments (planned and existing) and/or combination of experiments that are most applicable to validation of computational methods used for burnup-credit safety analyses.

VII. BURNUP CREDIT ANALYSIS SEQUENCE

ISG8 highlights the need for applicants employing burnup credit in criticality safety assessments to account for the axial and horizontal variation of the burnup within a spent fuel assembly. In practice, the axial burnup variation (e.g., the axial burnup profile) is commonly modeled in a criticality calculation using a finite number of axial segments or zones (10 to 20 is typical) to represent the burnup profile, each zone having a uniform average burnup for that segment. Consequently, implementation of burnup credit using this approach requires separate fuel depletion calculations for each axial zone, and the subsequent application of these spent fuel compositions in the criticality safety analysis. Implementation of this approach requires that numerous spent fuel depletion calculations must be performed, and potentially large amounts of data that must be managed, converted, and transferred between the depletion and criticality codes.

To simplify this analysis process and assist the NRC staff in their review of criticality safety assessments of transport and storage casks that apply burnup credit, a new SCALE control sequence, STARBUCS (**St**andardized **A**nalysis of **R**eactivity for **Bu**rnup **C**redit using **S**CALE) has been created.[16] STARBUCS automates the generation of axially-varying isotopic compositions in a spent fuel assembly, and applies the assembly compositions in a three-dimensional (3-D) Monte Carlo analysis of the assembly in a cask environment. The STARBUCS control sequence uses the new ORIGEN-ARP methodology[17] of SCALE to perform automated and rapid depletion calculations to generate spent fuel isotopic inventories in each axially-varying burnup zone of a fuel assembly. The analyst need only specify the average assembly irradiation history, the axially varying burnup profile, the actinides and, optionally, the fission products that are to be credited in the criticality analysis. An arbitrary number of axial zones may be employed, or the user may select from several pre-defined profiles. This series of calculations is used to generate a comprehensive set of spent fuel nuclide compositions for each axial zone of the assembly. The STARBUCS sequence uses the SNF inventories provided for each zone to automatically prepare cross sections for the criticality analysis. A 3-D KENO V.a criticality calculation is performed using cask geometry specifications provided by the user. Isotopic correction factors (ICFs) may also be applied to correct the criticality calculation for known bias and/or uncertainty in the prediction of the isotopic concentrations.

This new STARBUCS sequence has been used at ORNL to support the study of the impact of various assumptions that might be applied in the development of a loading curve. Figure 3 illustrates three loading curves highlighted against the 1998 inventory of U.S. discharged fuel. The loading curves show how the assumptions relative to selected nuclides and associated ICFs can lead to significant increases in the spent fuel inventory that can be loaded in a burnup credit cask. The curves indicate that, as discharge burnups and initial enrichments increase, efforts to incorporate fission products and/or reduce the ICFs will be needed to assure a burnup credit cask can carry a significant portion of the fuel anticipated for future discharge.

VIII. SUMMARY

The technical bases needed to help improve and expand the U.S. regulatory guidance for burnup credit in transportation casks have been developed at ORNL under the direction of the U.S. NRC research staff. *The goal has been to develop criteria and/or recommendations that are technically credible, practical, and cost effective while maintaining needed safety margins*. The technical work performed at ORNL is now undergoing final review by NRC staff and it is anticipated that changes to the recommendations of ISG8 will be forthcoming.

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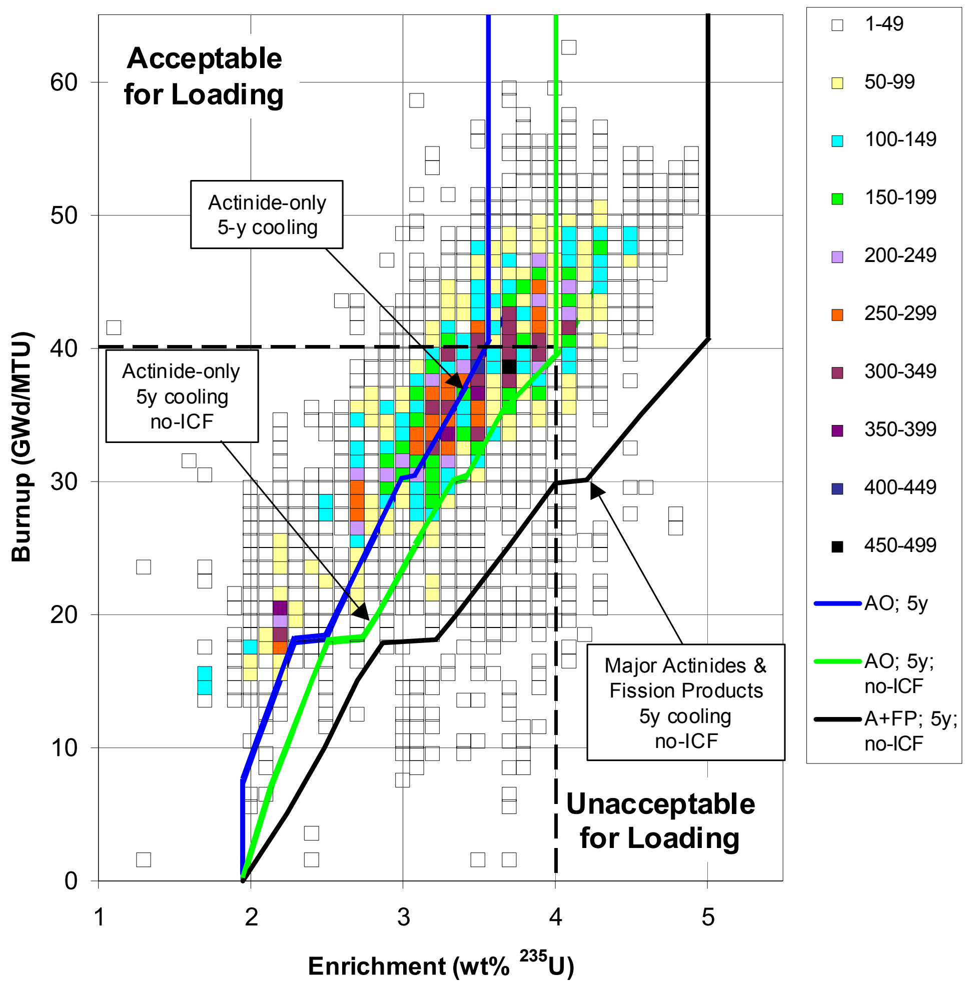

Figure 3. Illustrative loading curves for GBC-32 cask shown with PWR SNF discharge data through 1998 (numbers in legend indicate number of assemblies). Dashed lines represent current burnup and enrichment limits of ISG8.

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