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Research Supporting Implementation of Burnup Credit in the Criticality Safety Assessment of Transport and Storage Casks

C. V. Parks, I. C. Gauld, J. C. Wagner, B. L. Broadhead, M. D. DeHart
Oak Ridge National Laboratory*
P.O. Box 2008
Oak Ridge, TN 37831-6370

D. D. Ebert
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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Oak Ridge National Laboratory

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Abstract

The Nuclear Regulatory Commission (NRC) Office of Regulatory Research (RES) has initiated a program to support effective implementation of burnup credit in the criticality safety assessment of transport and dry storage casks. The goal is to develop technical bases that can be used to provide criteria and guidance for use in licensing activities. The program is being conducted in a phased approach, with the initial focus on unresolved issues related to the use of actinide-only burnup credit in transport and dry storage casks designed for spent fuel from pressurized-water reactors (PWRs). The work will gradually expand to investigate credit for fission products in PWR casks and application to burnup credit for boiling-water-reactor (BWR) fuel. This summary will review the status of progress to date and will identify planned activities and priorities.

Introduction

In the past, criticality safety analyses for commercial light-water-reactor (LWR) spent fuel storage and transport canisters assumed the spent fuel to be fresh (unirradiated) fuel with uniform isotopic compositions corresponding to the maximum allowable enrichment. This "*fresh-fuel assumption*" provides a well-defined, bounding approach to the criticality safety analysis that eliminates all concerns related to the fuel operating history, and thus considerably simplifies the safety analysis. However, because this assumption ignores the decrease in reactivity as a result of irradiation, it is very conservative and can result in a significant reduction in spent nuclear fuel (SNF) capacity for a given package volume. *The concept of taking credit for the reduction in reactivity due to fuel burnup is commonly referred to as burnup credit.* Numerous publications have demonstrated that increases in SNF cask capacities from the use of burnup credit can enable a reduction in the number of casks and shipments, and thus have notable economic benefits.

The use of burnup credit in criticality safety analyses for away-from-reactor applications (transport and storage) necessitates that the reactor operating history and conditions experienced by the fuel be considered. In contrast to the fresh fuel assumption, the use of burnup credit requires additional validation of calculational methods used to predict the SNF nuclide compositions applied in the safety analyses. Studies performed in the United States (sponsored largely by the Department of Energy and the Electric Power Research Institute) and abroad (primarily France, the United Kingdom, and Japan) have provided a significantly advanced understanding of the issues and developing approaches for a safety evaluation. However, a consensus has not been reached on how to answer such questions as: What constitutes adequate validation per the guidance of the ANSI/ANS-8.1 standard for nuclear criticality safety outside reactors? How does one select the appropriate axial-burnup profile for the licensing analysis? How should the variation/uncertainty in operating histories, fuel design, and SNF composition be quantified and incorporated in the safety analysis? The NRC is seeking to develop and document technical bases for criteria and guidance that will facilitate the review of licensing applications that use burnup credit. Such technical bases will allow the identification of areas where additional understanding or experimental information can enhance the safe and effective use of burnup credit.

The goal of the NRC/RES project directed at burnup credit is to develop technical bases and to provide recommendations on criteria and guidance for consideration by the licensing office (Office of Nuclear Material Safety and Safeguards, NMSS). The purpose of this paper is to review the progress of the NRC/RES efforts since its inception in the summer of 1999 and discuss ongoing and planned activities.

Current NRC Guidance on Burnup Credit

One of the initial activities of the NRC/RES project on burnup credit was to provide the NRC/NMSS Spent Fuel Project Office (SFPO) with confirmatory analyses and technical assistance to support the issuance of Revision 1 of the Interim Staff Guidance - 8 (ISG8),[1] which provides recommendations for the use of burnup credit with PWR spent fuel in transport and dry storage casks. A discussion of the technical considerations that helped form the development of ISG8 can be found in Ref. 2.

The recommendations within ISG8 limit the burnup credit to that available from actinide-only nuclides for SNF with an assembly-average burnup of 40 GWd/t or less and a cooling time of 5 years. The ISG8 recommendations allow spent fuel with burnup values greater than 40 GWd/t to be loaded in a cask, but burnup to only 40 GWd/t can be credited in the safety analysis. Initial enrichments up to 5.0 wt % ^{235}U are allowed but, for each 0.1 wt % increase above 4.0 wt %, the assigned burnup loading value must be 1 GWd/t higher than the credited burnup used in the safety analysis. This loading offset accounts for the lack of assay data for fuels with an initial enrichment greater than 4 wt %. In addition, assemblies with burnable absorbers are not allowed. The ISG8 recommends that the analysis methods used to predict the SNF isotopics and the neutron multiplication factor (k_{eff}) for the cask be validated against measured data. Potential uncertainties caused by a variation in reactor operating histories, a lack of measured data for validation, and a spatial variation of the burnup within the assembly (axial and horizontal) need to be quantified and/or bounded in the safety analysis. Further, ISG8 recommends the use of a measurement prior to or during the loading procedure to ensure that each assembly is within the loading specifications for the approved contents (e.g., a burnup measurement). The recommendations for a bounding approach and preshipment measurements are consistent with the international regulations for the transport of fissile material.

Although ISG8 does not recommend that credit be sought for the presence of fission products, it is recommended that applicants provide an estimate of the cask-specific reactivity margin provided by the fission products and actinides for which the computational methods cannot be adequately validated. It is recommended that the methods used for such estimations be verified against any available experimental data and/or computational benchmarks to demonstrate the performance of the applicant’s methods in comparison with independent methods and analyses.

A key element of ISG8 is the recognition that the "staff will issue additional guidance and/or recommendations as information is obtained from its research program on burnup credit and as experience is gained through future licensing activities." No commercial LWR cask has been licensed for burnup credit in the United States. ISG8 represents an initial step towards regulatory guidance that enables industry to effectively proceed with design and licensing of a burnup-credit cask. The goal of the research program is to provide information that can serve as a basis for decisions on potential future modifications to the ISG8 recommendations. Such future modifications should lead to enhanced usage of burnup-credit casks while maintaining an adequate margin of safety.

Overview of Current Research Efforts

With the criteria and guidance of ISG8 established by the licensing staff, the effort of the research project shifted to identifying work needed to develop expanded guidance relative to selected elements of ISG8, to implement software enhancements that can facilitate the use of computational methods in safety analyses, and to develop the technical basis for the NRC/SFPO to use in considering future revisions of ISG8. A baseline report[3] was developed to review the status of burnup credit and to provide a strawman prioritization for areas where additional guidance, information, and/or improved understanding were judged to be beneficial to the effective implementation of burnup credit in transport and dry storage casks. The prioritization considered input obtained at public workshops sponsored by the NRC and Nuclear Energy Institute (NEI).

As a result of the initial review and input from industry and licensing staff, the current focus areas for the NRC research program were established:

1. Development of a comprehensive reference report that uses current cask designs (rail and truck) to provide a consistent basis for demonstrating the magnitude of the various negative reactivity components as a function of burnup, initial enrichment, and cooling time.

2. Development of an automated process for coupling the depletion/decay analysis to the criticality analysis to support initial license reviews. Eventually the analysis tool will be released as a module of the SCALE code system.[4]

3. Development of a computational benchmark for a generic rail cask design to support the calibration of an applicant's estimation of fission product margin per ISG8 recommendations.

4. Development of criteria and guidance for the selection of an appropriate axial profile for use in the safety assessment.

5. Development of an initial recommendation and associated technical basis for potential near-term modifications to the ISG8 relative to the use of cooling times other than 5 years.

6. Development of an initial recommendation and associated technical basis for potential near-term modifications to the ISG8 relative to the use of burnup credit with PWR fuel containing burnable poison rods and/or integral burnable absorbers.

7. Investigation of the potential for modifying or removing the loading offset (the added burnup margin required for fuel with initial enrichments above 4.0 wt %) based on existing and potential experimental data.

8. Review and evaluation of existing and proposed experimental data to (a) demonstrate and rank the relevance of experiments for methods validation using quantitative criteria, (b) identify experimental

needs, and (c) assess technical bases for "certifying" a minimum reactivity margin accountable to fission products.

The first two of these eight areas are aimed at assisting the licensing staff in preparation for review of applications which use burnup credit in the safety analysis. Areas 3 and 4 are being developed to provide additional guidance to assist in effective implementation of the ISG8 recommendations. The remaining areas are directed at expanding the inventory of fuel that will be allowed in a burnup-credit cask. Progress in each of these eight areas will be discussed in the following sections.

Reference Report on Components of Negative Reactivity

A significant number of domestic and international studies have been performed to help understand the components that contribute to the negative reactivity available with burnup credit. However, most of these studies were not comprehensive in nature and a comparison between studies is often difficult due to different assumptions used in the analysis (e.g., nuclide sets, cask models, etc.) A study is being performed to provide the NRC with a comprehensive reference report that uses a consistent set of assumptions to demonstrate the components of negative reactivity as a function of initial enrichment, burnup, and cooling time. Two cask configurations have been utilized: a generic burnup-credit rail cask model with 32 PWR spent fuel assemblies (GBC-32), as shown in Fig. 1, and a truck cask model (not shown). The negative reactivity provided by various nuclide sets are considered. The goal of the report is to quantify the various contributors to the negative reactivity available in SNF and to provide explanations that will assist readers in understanding the physics associated with the various effects. As an example of the type of information to be presented and discussed in the report, consider the nuclide sets of Table 1 and the results shown in Figs. 2B4 for the GBC-32 cask configuration. These figures demonstrate the reactivity decrease from various nuclide data sets as a function of burnup and cooling time.

Burnup-Credit Analysis Sequence

The 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 (i.e., 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 therefore requires that numerous spent fuel depletion calculations must be performed, and potentially large amounts of data must be managed, converted, and transferred between the various 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. 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 assemblies in a cask environment.

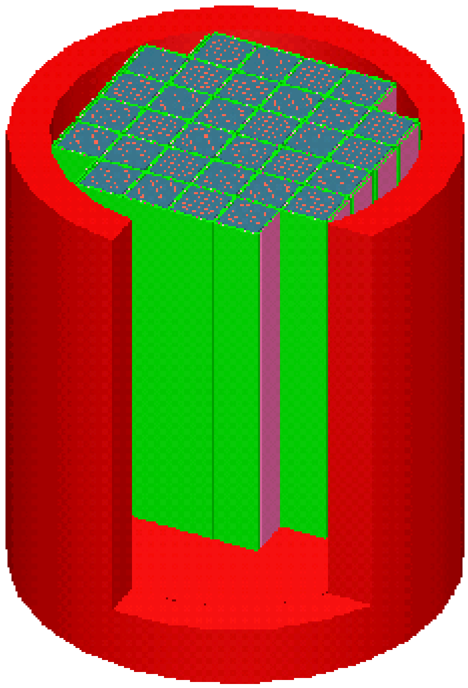

Figure 1. Cutaway view of GBC-32 burnup-credit cask model (one-half full height).

Table 1. Nuclide sets used for analysis

SET 1: Major actinides* (10 total)									
U-234	U-235	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	O†
SET 2: Minor actinides and major fission products (19 total)									
U-236	Am-243	Np-237	Mo-95‡	Tc-99	Ru-101‡	Rh-103‡	Ag-109‡	Cs-133	Sm-147
Sm-149	Sm-150	Sm-151	Sm-152	Nd-143	Nd-145	Eu-151‡	Eu-153	Gd-155	

*Actinides are consistent with those specified in the DOE Topical Report (Ref. 5).

†Oxygen is neither an actinide nor a fission product, but is included in this list because it is included in the calculations.

‡Nuclides for which measured chemical assay data are not currently available in the United States.

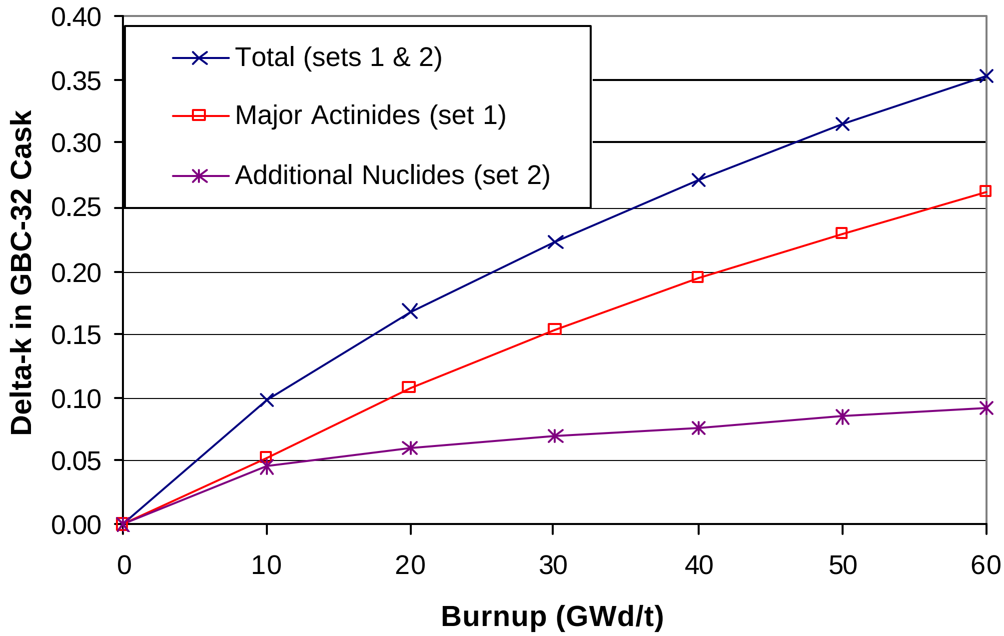

Figure 2.) k values (relative to fresh fuel) in the GBC-32 cask as a function of burnup using the different nuclide sets and 5-year cooling time for fuel of 4.0-wt % ^{235}U initial enrichment.

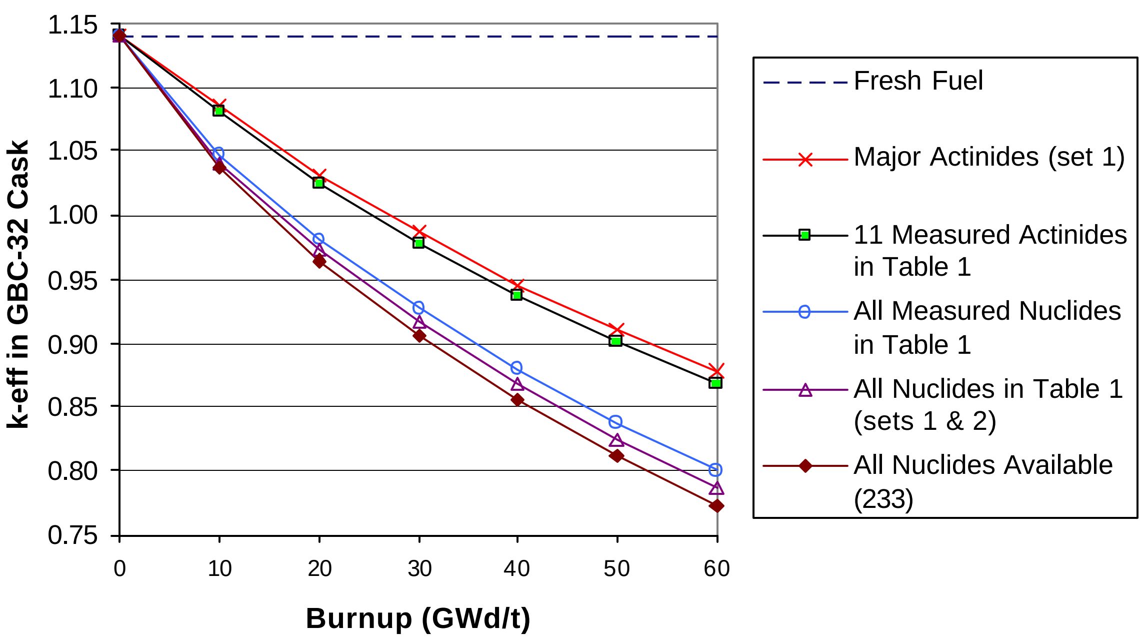

Figure 3. Values of k_{eff} in the GBC-32 cask as a function of burnup using various nuclide sets and 5-year cooling time for fuel of 4.0-wt % ^{235}U initial enrichment.

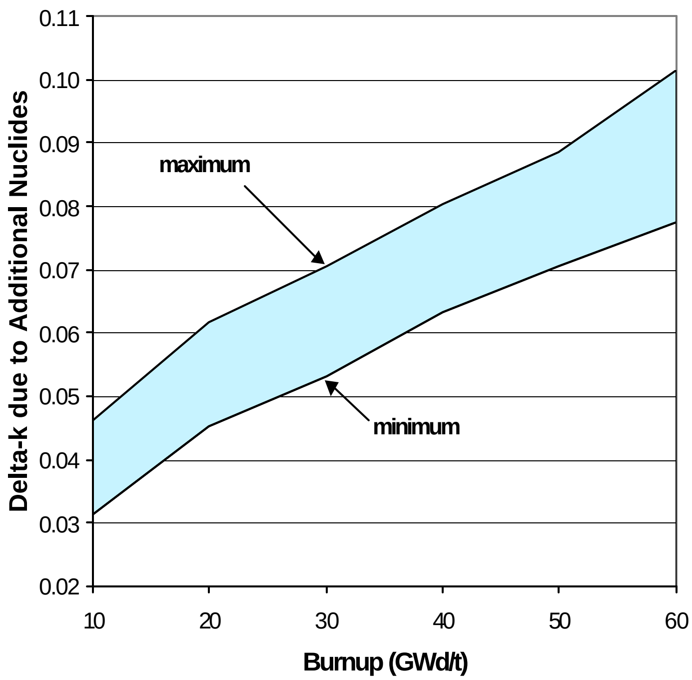

Figure 4. Range of Δk values in the GBC-32 cask due to the additional nuclides (set 2, defined in Table 1) as a function of burnup for all cooling times and initial enrichments considered (cooling times from 0 to 40 years; initial enrichments from 2.0 to 5.0 wt %).

The STARBUCS control sequence uses the new ORIGEN-ARP methodology 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 user 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. The user may input a profile or select from several predefined profiles. This series of calculations is used to generate a comprehensive set of spent fuel 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 may also be applied to correct for known bias and/or uncertainty in the prediction of the isotopic concentrations. Development of STARBUCS is ongoing and will include the ability to model horizontal burnup effects and may include an automated source-starting routine for the Monte Carlo criticality safety calculation to help ensure proper source convergence within a reasonable number of neutron histories.

Computational Benchmark for Estimation of Additional Reactivity Margin

A recommendation of ISG8 suggests that an applicant estimate the design-specific reactivity margin provided by fission products and actinides that are excluded from the safety analysis. The applicant is encouraged to assess this estimated margin against estimated uncertainties and/or potential nonconservative approximations that are not readily quantified. The ISG8 points to the small amount of experimental data available and computational benchmarks as a means to verify the estimate of the additional reactivity margin. The NRC research program has worked to develop and document a computational benchmark problem based on the GBC-32 model of Figure 1. While preserving design features common to current storage and transport cask designs and deemed important to the neutronic analysis, the benchmark problem approximates (or eliminates) nonessential detail and is not constrained by proprietary information. Thus, the benchmark provides a reference configuration that applicants can readily use to evaluate their estimate of the reactivity margin from nuclides not included in their explicit validation. Version 4.4a of the SCALE code system has been used to provide reference estimations of additional reactivity margin as a function of initial enrichment, burnup, and cooling time. Although the reference solutions are not directly or indirectly based on experimental results, the SCALE 4.4a system is being used to analyze available assay data and proprietary reactivity worth experiments to obtain partial validation of this particular methodology.

Guidance for Selection of Axial Profile

The ISG8 recommends the use of analyses that provide an "adequate representation of the physics" and notes particular concern with the axial and horizontal variation of the burnup. The horizontal variation of burnup is a relatively minor effect which has been investigated only within the context of the development of Ref. 5. Future work of the research program will seek to further investigate the horizontal variation of the burnup. However, the axial burnup profile is an extremely important component of the safety analysis. The profile is dependent on the fuel assembly design, burnup, and the operating conditions of the reactor. Work sponsored by the DOE has provided a database[6] of more than 3000 PWR axial-burnup profiles, and studies[7] have identified the axial profiles that provide bounding k_{eff} values over selected burnup ranges and developed artificial bounding profiles over select burnup ranges. The database provides a large, but not exhaustive, set of profiles that hopefully represents a statistical sampling of typical and atypical profiles resulting from irradiation in U.S. PWR reactors. Figure 5 shows the spread of k_{inf} values that result from the set of profiles available from a selected burnup range, together with the bounding "real" (i.e., actual profile from the database) and "artificial" profiles. Although applicants will always have the flexibility to extend the existing database and/or create and use alternative databases, the current research program is seeking to initially develop a technical basis which demonstrates that bounding profiles developed from the existing publicly available database are adequate for use in burnup-credit safety analyses with actinide-only assumptions. The technical basis for this recommended position is currently being documented for review by NRC staff.

As evidenced from Fig. 5, the use of a bounding profile provides a considerable increase in reactivity over the predominant "typical" or average profiles. Future work will seek to use risk-informed insights to enable criteria for the development and use of an "average" profile. For example, if axial-profile measurements for each assembly were performed prior to loading, a profile deemed bounding of the "typical" profiles could be used in the safety analysis and the profile for the as-loaded assembly would be checked for adherence. However, alternative approaches to allow the use of an average profile without such axial measurements are being investigated.

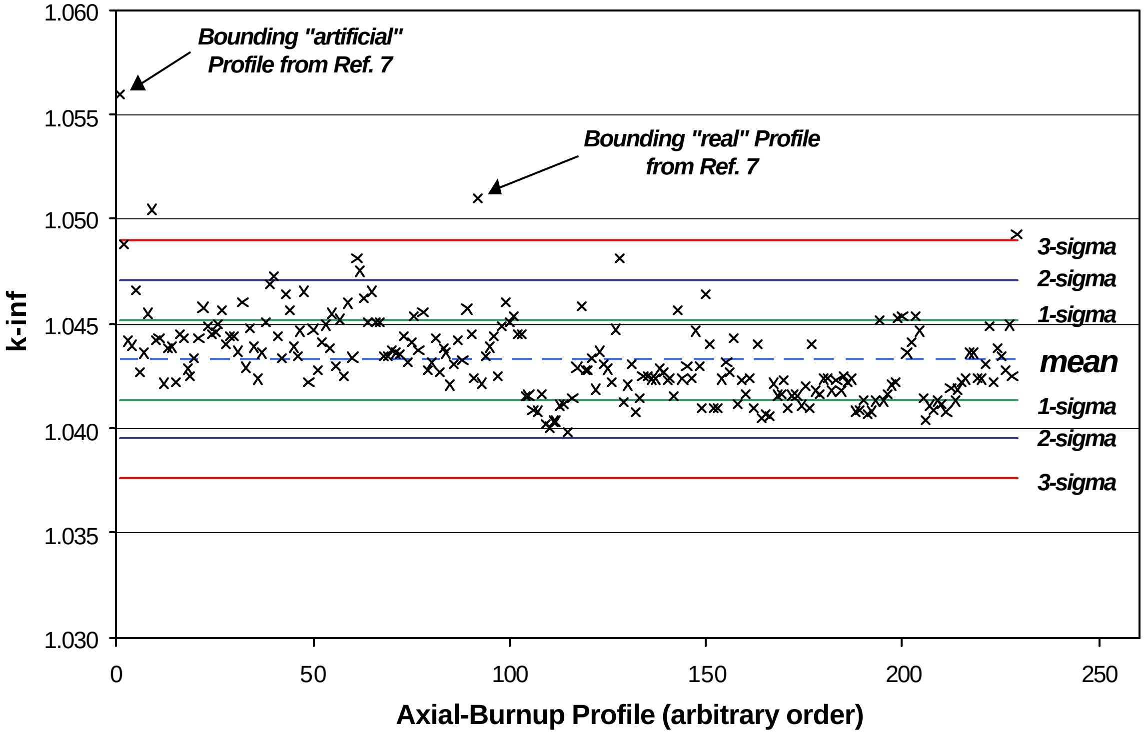

Figure 5. k_{inf} values based on database axial profiles for 38–42 GWd/t.

Modification of Cooling-Time Recommendation

The ISG8 recommends that a fixed cooling time of 5 years be used in the criticality safety analysis and that fuel with less cooling not be loaded in a burnup credit cask. In response to industry comments requesting more flexibility, work has been performed to demonstrate the effect of cooling time from discharge to 100,000 years and make recommendations relative to allowing additional cooling times. Figure 6 illustrates the general variation in k_{eff} as a function of cooling time for the GBC-32 cask loaded with SNF at 4.0-wt % initial enrichment and 40-GWd/t burnup. The "dip" at around 100 years is due to the decay of ^{241}Pu and the buildup of ^{241}Am and becomes less pronounced as the burnup decreases for a constant initial enrichment (i.e., under-burned fuel). For burnup-credit criticality safety analyses performed at 5 years, increasing cooling time results in an increasing conservative safety margin out to approximately 100 years. The magnitude of the conservatism depends on the initial enrichment and burnup of the fuel. Uncertainty associated with reactivity changes due to cooling time in the 1-to-100 year time period should be small because decay data important to the changes in this time period are known with very good accuracy.

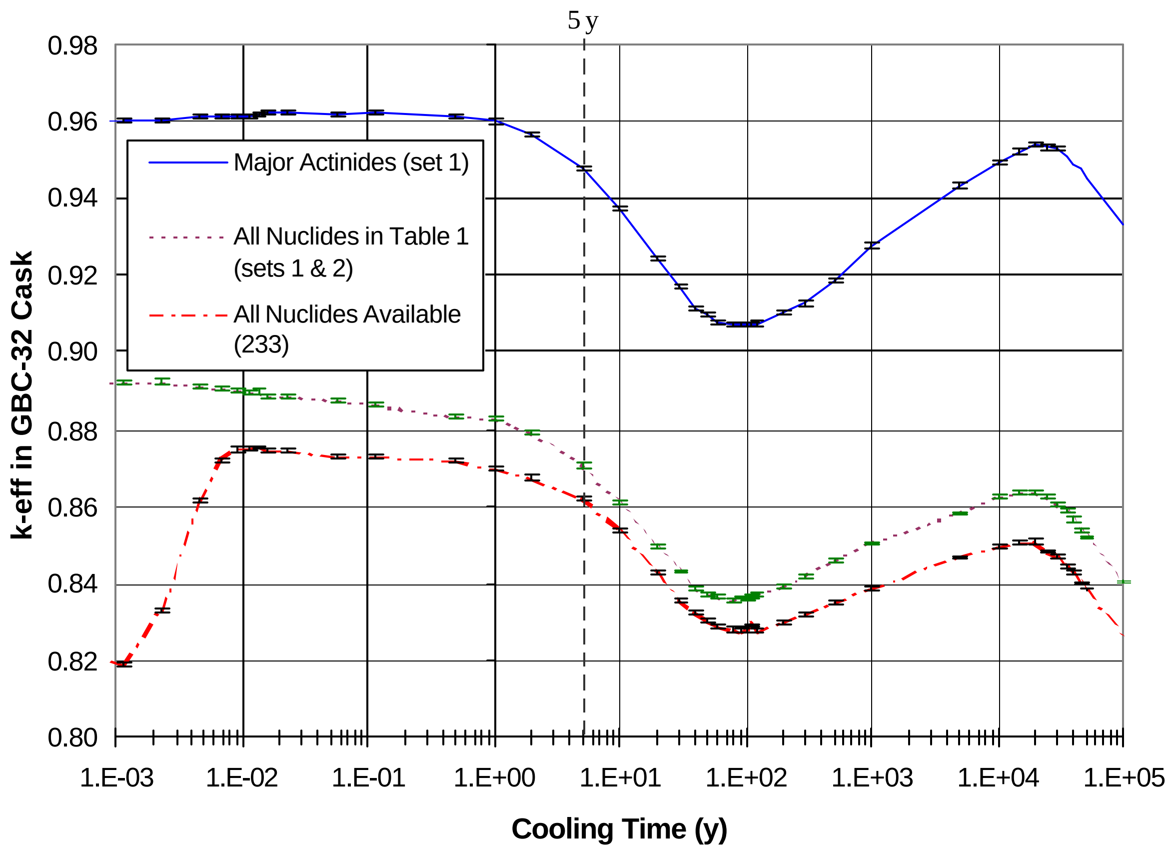

Figure 6. Values of k_{eff} in the GBC-32 cask as a function of cooling time for the three classifications of burnup credit (burnup-credit classifications are defined in Table 1). The results correspond to fuel with 4.0-wt % ^{235}U initial enrichment that has accumulated a 40-GWd/t burnup.

As evidenced by Fig. 7, there is an insignificant benefit in performing a safety analysis with cooling times greater than 50 years. 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. However, if SNF loaded with an assumed cooling time of 40 years remains in the cask beyond the 200-year time frame, then the potential may exist for a reactivity increase beyond that allowed in the safety assessment. A study of the reactivity margin provided by the actinide-only assumption could be used to dispense with this concern. To address this concern and lay a consistent foundation that enables future extension beyond the actinide-only assumption, it has been suggested that a value of 10 years be assumed as the cooling time limit for safety analysis. The rationale is that, except for SNF that is highly under-burned (e.g., 5.0 wt %, 20 GWd/t), 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). Finally, a lower limit on cooling time will continue to be dictated by thermal and shielding requirements.

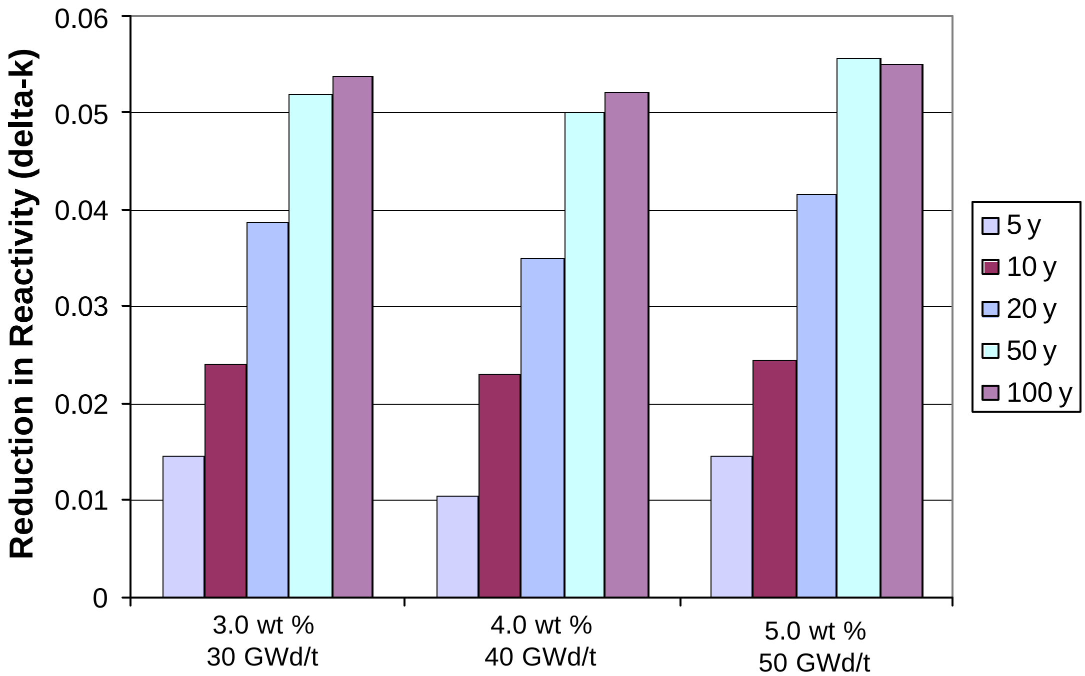

Figure 7. Reactivity reduction as a function of cooling time for some typical initial enrichment and discharge burnup combinations with actinide-only burnup credit.

Modifications to Allow SNF Exposed to Burnable Absorbers

The ISG8 restricts the use of burnup credit to assemblies that have not contained burnable absorbers during any part of their exposure. This restriction eliminates a large portion of the currently discharged spent fuel assemblies from cask loading, and thus severely limits the practical usefulness of burnup credit. Burnable absorbers may be classified into two distinct categories: (1) burnable poison rods (BPRs) and (2) integral burnable absorbers. BPRs are rods containing neutron-absorbing material that are inserted into the guide tubes of a PWR assembly during normal operation and are commonly used for reactivity control and enhanced fuel utilization. Due to the depletion of the neutron-absorbing material, BPRs are often (but not always) withdrawn after one-cycle residence in the core. In contrast to BPRs, integral burnable absorbers refer to burnable poisons that are a nonremovable or integral part of the fuel assembly. An example of an integral burnable absorber is the Westinghouse Integral Fuel Burnable Absorber (IFBA) rod, which has a coating of zirconium diboride (ZrB_2) on the fuel pellets.

The presence of BPRs during depletion hardens the neutron spectrum because of the removal of thermal neutrons by capture in ^{10}B and by displacement of moderator, resulting in lower ^{235}U depletion and higher production of fissile plutonium isotopes. Enhanced plutonium production and the concurrent diminished fission of ^{235}U due to increased plutonium fission have the effect of increasing the reactivity of the fuel at discharge and beyond. Consequently, an SNF assembly exposed to BPRs will have a higher reactivity for a given burnup than an assembly that has not been exposed to BPRs.

SCALE/SAS2H depletion calculations were performed assuming the BPRs were present during (1) the first cycle of irradiation, (2) the first two cycles of irradiation, and (3) the entire irradiation period (i.e., three cycles). For comparison purposes, isotopics were also calculated assuming no BPRs present. These four sets of isotopics were then used to determine the reactivity effect of each BPR design as a function of burnup for out-of-reactor conditions at burnup steps of 1 GWd/t and zero cooling time. The criticality calculations were based on an infinite array of spent fuel pin cells using isotopics from the various BPR depletion cases, and thus the effect of the BPRs is determined based on their effect on the depletion isotopics alone (i.e., the BPRs are not included in the criticality models).

Figure 8 plots the reactivity differences ( ) k values relative to no-BPR depletion calculations) as a function of burnup using the actinides from Table 1. The isotopics used in the criticality calculations correspond to spent fuel with 4.0-wt % ^{235}U initial enrichment that has been exposed to Westinghouse Wet Annular Burnable Absorber (WABA) rods during depletion. For the purpose of the depletion calculations, three cycles of 15-GWd/t burnup per cycle were assumed. The results shown in Fig. 8 demonstrate that the reactivity effect increases with BPR exposure (burnup and number of BPRs present) and that calculations based on continuous exposure during the entire depletion yield higher (more conservative) reactivity than analyses based on actual/typical one-cycle exposures. For the same conditions plotted in Fig. 8, but with the inclusion of the major fission products, the reactivity behavior is very similar to that of the actinide-only condition.

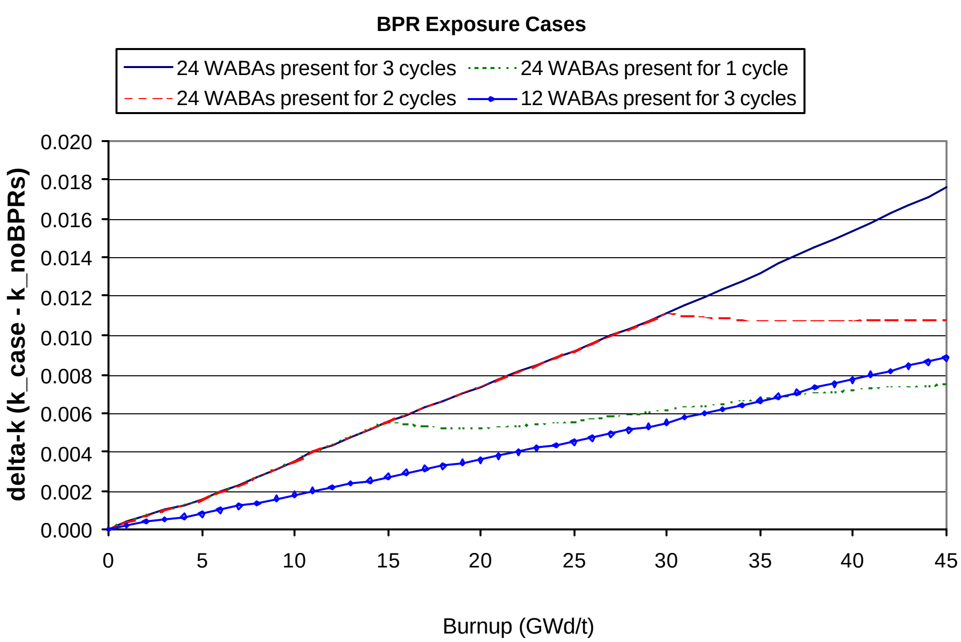

Figure 8. Reactivity differences ( ) k values relative to the no-BPR condition) as a function of burnup for various BPR exposures using actinide-only assumption. Results correspond to fuel with 4.0-wt % ^{235}U initial enrichment that has been exposed to Westinghouse WABA rods (three cycles of 15-GWd/t burnup per cycle were assumed).

Analysis of the GBC-32 cask loaded with Westinghouse 17 × 17 OFA assemblies provides k_{eff} values for actinide-only and actinide + fission-product-burnup credit that demonstrate a BPR effect very similar to that exhibited for an infinite array of fuel pins. To determine the impact of incorporating the axial-burnup distribution, k_{eff} values were also calculated for the GBC-32 cask for various BPR exposures with the axial-burnup distribution included. The results reveal that the inclusion of the axial-burnup distribution reduces the reactivity increase associated with the BPRs. This reduction is due to the fact that the lower-burnup regions near the ends (that control the reactivity of the fuel when the axial-burnup distribution is included) have less burnup, and thus less-than-average burnup exposure to the BPRs.

A SAS2H fuel assembly model is limited to a one-dimensional radial model with a single smeared fuel region. Geometric modeling approximations are made in an effort to achieve a reasonable assembly-average neutron energy spectrum during the depletion process. However, the presence of BPRs challenges SAS2H modeling capabilities. Therefore, HELIOS, a two-dimensional, generalized-geometry transport theory code was utilized for selected cases to compare the reactivity differences (Δk values relative to the no BPR condition) as a function of burnup against those established using SAS2H. The results were comparable within a few tenths of a percent, with SAS2H isotopics predicting slightly larger reactivity effects. Further, very good agreement was achieved between k_{inf} values based on isotopics from the two methods.

The reactivity effect of BPRs increases nearly linearly with burnup and is dependent upon the number and poison loading of the rods and the initial fuel enrichment. Although variations are observed for the various BPR designs, maximum reactivity increases have been found to be ~1 to 3% when maximum BPR loading and exposure time are assumed for typical initial enrichment and discharge burnup combinations. Based on the analysis summarized here, guidance for an appropriate approach for calculating bounding spent nuclear fuel isotopic data for assemblies exposed to BPRs may be developed. For example, assuming maximum BPR exposure during depletion would be a simple, conservative approach to bound the reactivity effect of BPRs, where maximum BPR exposure may be defined as the maximum possible number of BPRs with the most bounding BPR design (i.e., most bounding geometric design and maximum possible poison loading) for the entire exposure. Other, less-conservative approaches that incorporate information regarding the percentage of assemblies exposed to mutiple cycles will be explored during the coming year.

A study has recently been completed that investigated the impact of integral burnable absorbers on the k_{eff} values in cask environments. Depending on the design and loading of neutron poison, the presence of integral burnable absorbers can slightly lower or raise the k_{eff} values of SNF assemblies, in comparison to assemblies without the integral burnable absorber. Integral burnable absorber analyses for multiple designs have been studied, and the maximum increase in k_{eff} is less than that identified for assemblies exposed to BPRs. The technical basis for including assemblies with integral burnable absorbers is currently being documented for review by NRC staff.

Modification or Removal of Loading Offset

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/t). Only seven of the approximately 50 samples had BPRs present during irradiation. The enrichments and burnup ranges of the spent fuel samples used in recent validation studies performed for burnup-credit studies are shown in Fig. 9. The figure illustrates the paucity of experimental data in both the high-enrichment and high-burnup regimes. 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. The loading offset, expressed in terms of the reactivity penalty Δk, is illustrated in Fig. 10 for the GBC-32 cask design employing actinide-only burnup credit for a 5-year cooling time. The reactivity margin for 5-wt % fuel, the maximum enrichment considered by ISG8, ranges from typically 0.035 to 0.045 Δk, depending on the fuel burnup.

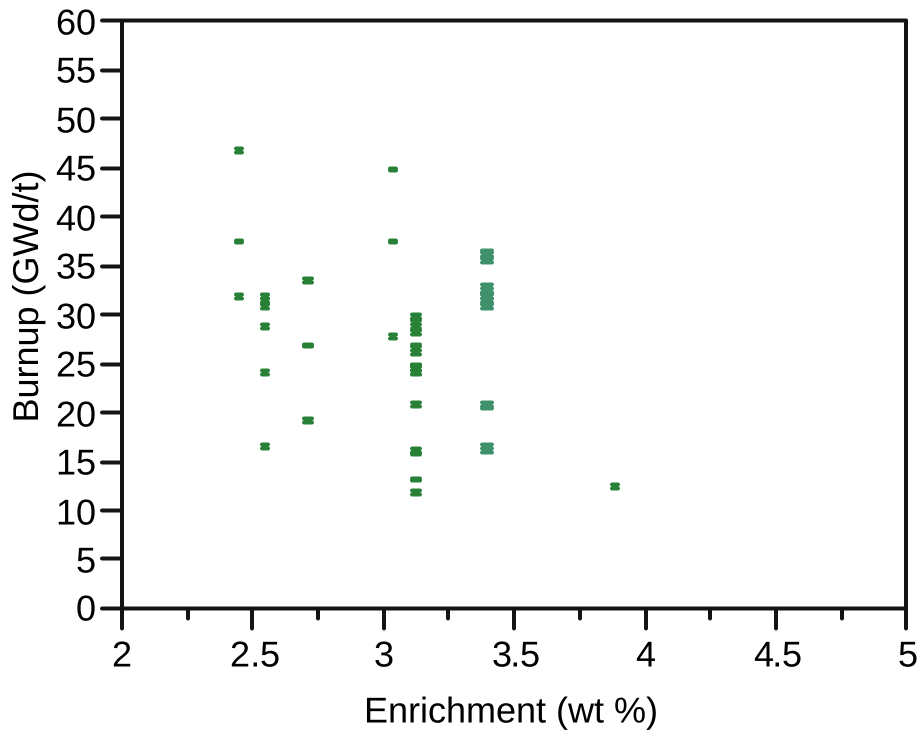

Figure 9. Enrichment and burnup of 46 PWR assay samples used in recent burnup-credit isotopic validation studies.

This added margin, shown in Fig. 10, can be compared with the actinide isotopic uncertainties for which it is intended to compensate as a means of estimating the conservatism in ISG8 with respect to existing isotopic assay data and spent fuel characterization methods. The influence of actinide uncertainties on the predicted k_{eff} of a spent fuel cask was estimated using isotopic correction factors derived from the publicly available experimental assay data obtained with the depletion analysis methods in SCALE and ENDF/B-5 cross-section data. The correction factors represent the amount by which the isotopic compositions must be adjusted to account for known calculational bias and uncertainty. This uncertainty is typically accounted for at a 95% confidence level and allows for the variance of the predicted bias and the number of assay measurements available. The influence of the uncertainties on the calculated k_{eff} was estimated using sensitivity coefficients that have been generated for each of the important actinides over a wide range of enrichments and burnup values. The sensitivity coefficients represent the relative change in k_{eff} with respect to a 1% change in a nuclide concentration. Thus, the coefficients provide a method of predicting the change in k_{eff} with respect to a change in the isotopic concentrations required to account for the uncertainties in the concentrations.

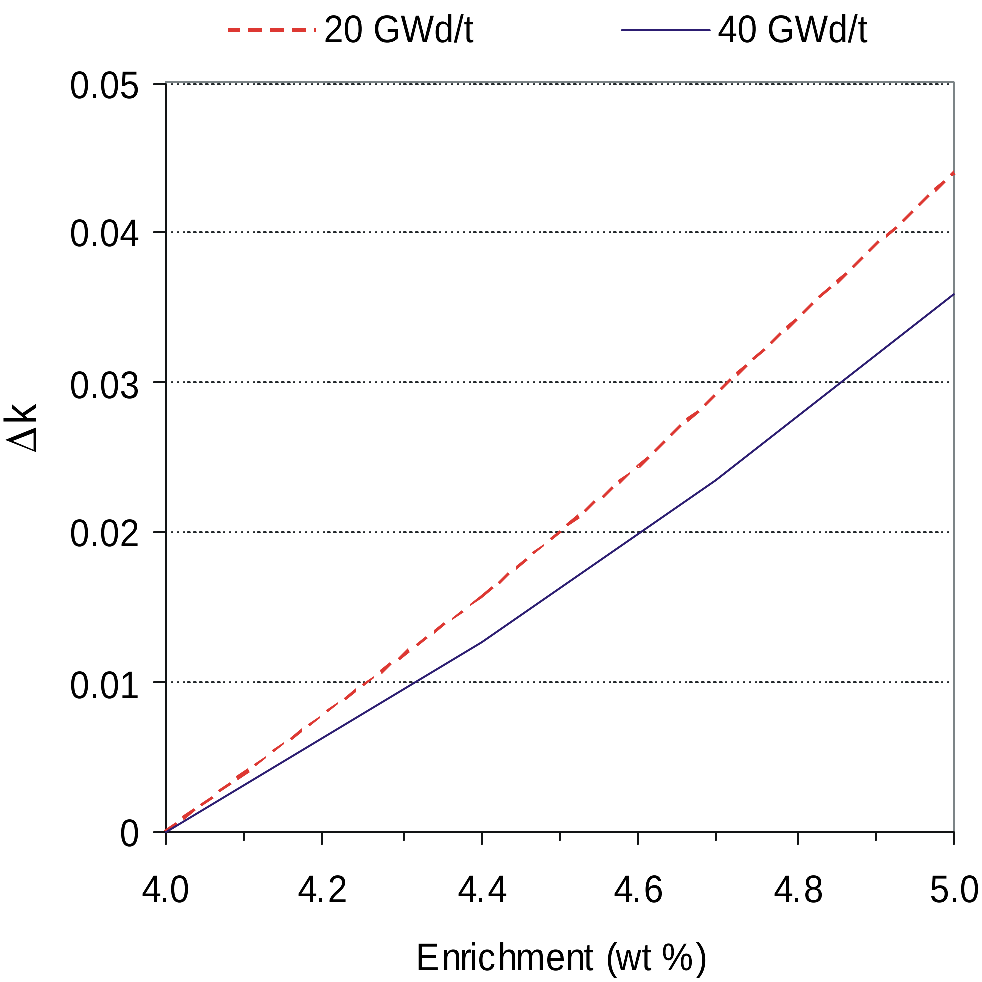

Figure 10. ISG8 loading offset reactivity for the GBC-32 cask design.

An important consideration is how to properly combine the uncertainties of the individual isotopes. The most conservative approach adjusts the concentration of every nuclide in such a way as to always create a more reactive system. Perhaps a more realistic strategy is to assume each uncertainty is independent (i.e., random) and combine the uncertainties using a root-mean-square (RMS) approach. However, the RMS method does not consider potentially correlated uncertainties in transmutation or decay chains. The actual effect is somewhere between these two approaches. Assuming the more conservative strategy, the net reactivity margin associated with the actinide uncertainties is illustrated in Fig. 11 for a range of enrichments and burnup. The figure shows the increase in the reactivity margin associated with uncertainties in the concentration of the dominant burnup-credit actinides with increasing burnup. The changes in the margin reflect the changing actinide compositions with burnup and enrichment, the bias and uncertainty associated with each actinide, and the changing relative importance of each actinide to the system reactivity. As enrichment increases, the overall uncertainty exhibits a marginal decrease. For high-burnup fuel the combined reactivity change associated with all actinide isotopic uncertainties is about 4 to 5% $\Delta k/k$. If the actinide uncertainties are combined using a less-conservative RMS approach, the margin is reduced considerably to about 2% $\Delta k/k$. The reactivity margin due to the isotopic uncertainties is considerably larger than that due to the average bias.

Figure 11 inherently assumes that the isotopic uncertainties do not change with increasing enrichment. That is, the isotopic correction factors derived using the existing database of lower-enrichment and moderate-burnup fuel are assumed to be applicable in the extended regimes. The ISG8 loading offset above 4 wt % (see Fig. 10) amounts to about an added 4%)k/k penalty (assuming a neutron multiplication factor near unity) for an enrichment of 5 wt %, a reactivity margin similar to that associated with current actinide uncertainties. Therefore, the ISG8 loading offset penalty (corresponding to 5.0 wt %) is approximately equivalent to doubling the isotopic correction factors derived using existing isotopic assay data below roughly 3.5 wt % and 40 GWd/t.

A number of new sources of experimental assay data have been identified that could potentially be used to assess isotopic bias for the higher-enrichment and higher-burnup regimes in the near term. Some assay data may become available from an experiment performed on 3.8-wt % high-burnup fuel in Japan. Although this fuel does not extend beyond 4 wt %, it would significantly improve the coverage by existing data and provide improved confidence in code predictions above 3.5 wt %, thus providing a potential basis to extend the range of applicability above 4 wt % using bias trends. Additional information on the reactor operating conditions is currently being pursued to enable their accurate analysis.

The most attractive sources of existing higher-enrichment data that have been identified are the proprietary French programs, primarily the Gravelines-3 program involving 4.5-wt % fuel with a wide range of burnup. Acquisition of these data is currently viewed as a high priority within the NRC research project, particularly with the exclusion of some data sets from future consideration due to the use of nonstandard (reconstituted) assemblies. Published differences between French calculations and experiments[8] indicate no significant trends with burnup for the major burnup-credit actinides and, notably, the magnitude of the calculated isotopic biases for the 4.5-wt % fuel are comparable to the biases observed in benchmarks in the U.S. studies involving lower-enrichment fuels. However, the French results were obtained using cross-section data from the Joint European Files (JEF) of evaluated data and two-dimensional depletion analysis methods. Consequently, the reported biases may not be indicative of different code systems and data. Nevertheless, the results suggest that with up-to-date nuclear data and appropriately rigorous computational methods the burnup-credit actinides can be predicted in high-enrichment and high-burnup PWR fuel to a level of accuracy that is not significantly different than that for conventional enrichment and burnup fuel.

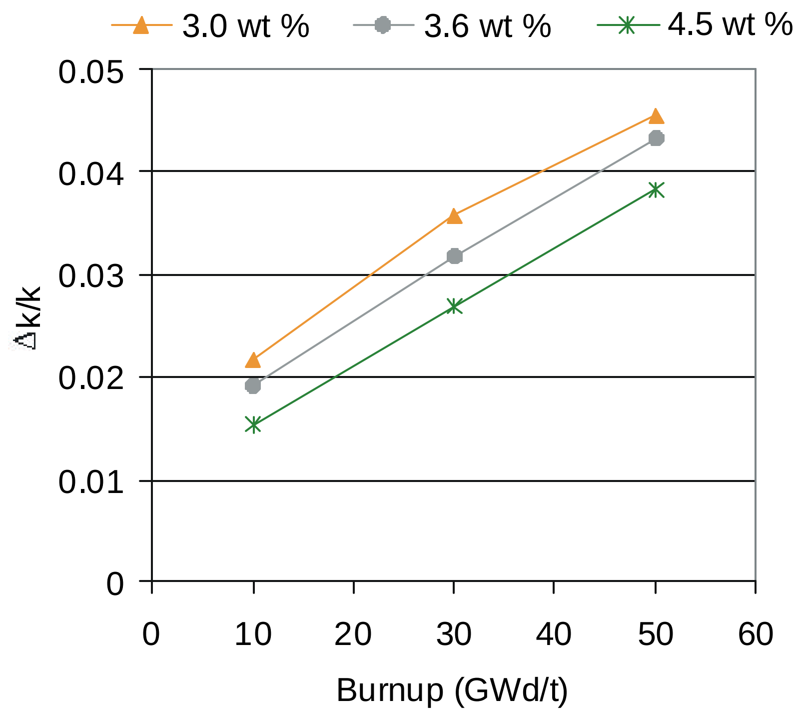

Figure 11. Reactivity penalty associated with actinide uncertainties as a function of burnup.

Isotopic analyses of spent PWR fuel will be performed as part of the REBUS program[9] and Phase II of the LWR-PROTEUS program.[10] These programs are proprietary. NRC is currently a participant in REBUS and will have access to the isotopic assay data. The projected schedule for reporting the commercial UO_2 radiochemical analysis results from REBUS is late 2002 to early 2003. The LWR-PROTEUS Phase II measurements are scheduled for completion in July 2001. The only new data likely to become available in the United States in the near term are the spent fuel samples from TMI-1 (4 and 4.65 wt %) currently being analyzed for the DOE repository program. However, release of the final results from this program are not expected before June 2001. The PWR enrichment and burnup regimes covered by these programs, the Japanese assay data, and the Gravelines-3 proprietary data are shown in Fig. 12, together with the existing publicly available assay data. A report providing a detailed review of available isotopic assay information and discussing the sensitivity methods being applied to investigate similarity and expected trends has been drafted and provided to NRC for review and comment.

Extending the area of applicability by making use of trends in the bias and uncertainty has proven to be challenging due to a relatively large variability in the existing data and the many factors that may influence the bias, including fuel enrichment, burnup, assembly design complexity, calculational methods, nuclear data, and uncertainties in reactor operating conditions, irradiation history, and sample burnup. A reliable trending assessment is challenged by the limited amount of experimental data and the large number of different parameters that can affect the bias.

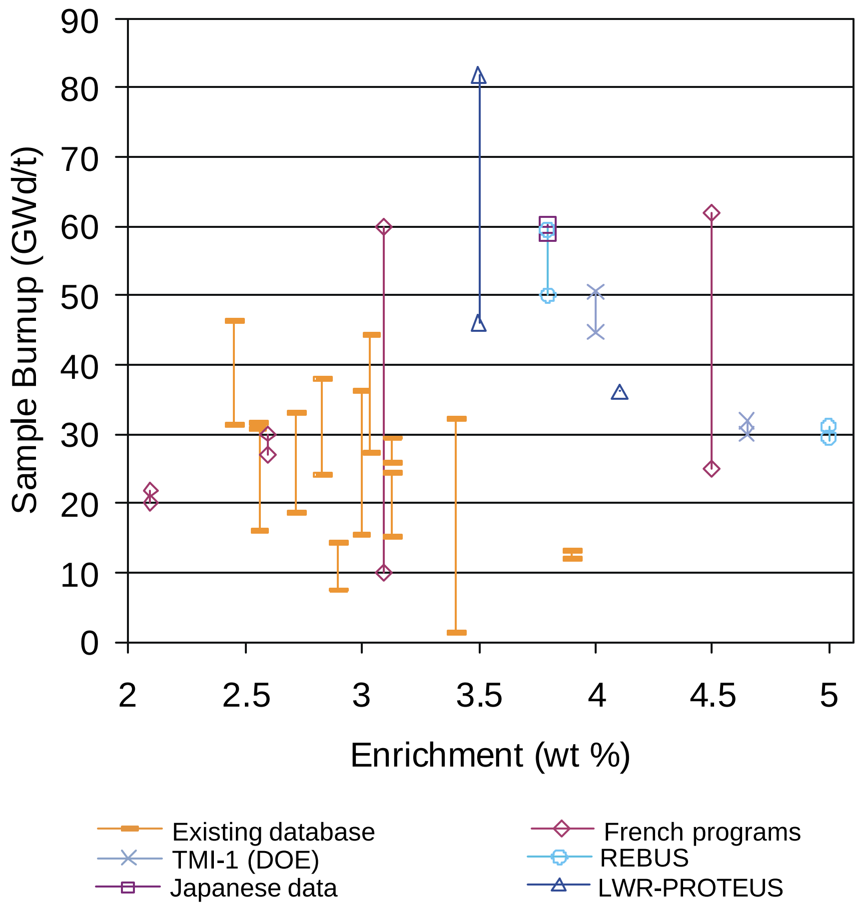

Figure 12. Available and potential future PWR isotopic assay data.

Several studies do suggest, however, that the effect of enrichment on the isotopic uncertainties should be minimal. The published French results 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 to assess the influence of nuclear data bias and uncertainties on the isotopic compositions and the k_{eff} of a spent fuel storage cask. 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. However, there is currently insufficient experimental data to validate these findings. It is anticipated that as new assay data become available it will be possible to combine the limited amount of experimental data with the sensitivity-based methods to provide additional evidence to support predictions beyond the range where the majority of experimental data exist.

Evaluation and Use of Experimental Data

The nature of experimental data appropriate for use in the validation of burnup-credit analysis methodologies and the value and applicability of such data have been debated topics for over a decade. Available (albeit some are proprietary) experimental data include chemical assays of SNF 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.

A summary of chemical assay data was discussed earlier in the section on the loading offset. Reactor-critical configurations and some planned reactivity worth experiments with spent fuel are integral experiments that require the prediction of the nuclide composition and k_{eff} analysis. Thus, these experiments are also potential sources of experimental data that may be used to supplement, or potentially replace, the use of assay data. These options will continue to be considered as the research project proceeds.

Currently the NRC/RES is actively participating in the REBUS experimental program[9] which will provide chemical assay data and reactivity worths from insertion of SNF rods into a matrix of fresh fuel pins. The NRC is also discussing with the French the various avenues available for potential use of portions of their critical experiment and chemical assay data. To assist in this assessment, sensitivity/uncertainty (S/U) methods discussed in Ref. 11 are being used to provide information on the strengths and potential limitations of various types of experiments relative to validation needs for burnup credit. The S/U methodology utilizes two different parameters as measures of applicability: one is a global measure for system-to-system applicability (c_k value); the other is a nuclide-specific measure of applicability (T value). Existing fresh fuel (UO_2-fuel and mixed-oxide) critical experiments, reactor- critical configurations, reactivity worth experiments, and measured chemical assay data are being studied with the prototypic S/U methods.

The experimental programs evaluated using S/U methods include the French critical experiments with HTC pins,[12] which simulate the actinide concentrations of burned spent fuel (37 GWd/t) without the presence of fission products, and the Valduc[13] fission-product-solution experiments in which fission products are individually placed into solutions at the center of an experimental core. Other experiments that are to be analyzed include the worth experiments from the CERES/MINERVE[14] program, the planned ANL(NRAD) program,[15] the REBUS program[9] in Belgium, the PROTEUS program[10] in Switzerland, and the planned DOE-sponsored program[16] at Sandia National Laboratories.

The c_k values for the HTC experiments indicate a high degree of applicability to a series of infinite pin-cell calculations for burnups ranging from 10 to 60 GWd/t. The T values also indicate a high degree of applicability for the primary plutonium isotopes for burnups less than 60 GWd/t. Thus, these experiments are believed to be beneficial to actinide-only burnup-credit validation efforts.

The Valduc fission-product experiments are evaluated using only the nuclide-specific T parameters. This method is used because the system-to-system parameters are not currently appropriate for fission products due to the lack of uncertainty data on the fission-product cross sections. Also, an examination of the T values is performed only for ^{149}Sm, because this is the only experiment available in the open literature. The T values obtained for the single Valduc experiment indicated that it is highly applicable to ^{149}Sm capture in the series of pin-cell applications for 10 to 60 GWd/t. This indicates that the fission-product-solution experiments should be good experiments for the validation of the fission products in a cask environment. These fission-product-solution experiments are valuable in that they allow for the effect of individual fission-product cross-section uncertainties on the system k_{eff} to be evaluated separately. This information is useful in combination with the additional data derived from the CERES-type measurements with doped-fission products, where the contribution of fission-product cross-section uncertainties to the worth of the fission product itself is obtained. The CERES-type measurements are very sensitive to the fission-product cross-section uncertainties; however, the Valduc-type experiments have fission- product cross-section sensitivities that are nearly the same as those in an actual cask environment. The details on the configurations for the REBUS, PROTEUS, and ANL(NRAD) are not currently available, and only approximate models have been considered. Further analyses are in progress.

Three PWR commercial reactor-critical state points have also been analyzed using the S/U methodology, and comparisons made with SNF cask environments. The results indicate that the reactor-critical state points have adequate similarity to cask environments. Reactor-critical configurations are the only measured information where significant quantities of SNF are used and, from an integral perspective, provide a viable source of validation information for both actinides and fission products.

A prudent approach to burnup-credit validation should involve assay-data validation, followed by cross-section validation for the actinides and fission products. The existing mixed-oxide fuel criticals, combined with French HTC experiments, are believed to be sufficient for actinide-only cross-section validation purposes. Additionally, applications that take credit for fission products need to consider experiments which validate individual fission-product cross sections. Validation is best accomplished by a combined approach of large-sample, individual fission-product worth measurements, like the Valduc or DOE/SNL experiments and the small-sample, individual doped-fission-product worth measurements like CERES/MINERVE. The remaining REBUS, PROTEUS, and ANL spent-fuel-sample worth experiments may be useful as an overall check on the reactivity effects of spent fuel. Although more complex to model, commercial-reactor-critical data provide a valuable source of experimental information for integral validation of the SNF compositions and cross sections and the effect of neutronic interaction between assemblies.

The estimation of the benefits of inclusion of fission products in the reactivity effects of burnup is a complicated process. An approach that has been offered is to quantify two independent factors to account for the effects of isotopic prediction inaccuracies and isotope cross-section inaccuracies. The product of these two factors and the predicted worth values in the cask configuration gives an estimate of the "guaranteed" fission-product worth in the cask application of interest. Efforts are underway to quantify these effects for an example application.

Expert Input and Review

The NRC research program is working to obtain input from domestic and international experts and organizations with experience in burnup-credit research, experiments, criticality safety practice, and operations of transport and dry storage casks. One primary tool for this input is the expert panel convened to participate in a process of developing Phenomena Identification and Ranking Tables (PIRT). The main goal of the PIRT panel is to identify phenomena, parameters, procedures, etc., that influence the determination of k_{eff} for spent fuel in a cask environment, provide a graded (e.g., high-importance, moderate-importance, low-importance) ranking of the phenomena and, as appropriate, judge the uncertainty associated with each phenomena. Besides its primary objective, the PIRT process can also facilitate a beneficial exchange of information and ideas that will hopefully lead to improved understanding of the issues and practical approaches for effective implementation of burnup credit within the licensing process. Issues related to axial profile, cooling time, presence of BPRs, and loading offset have all been presented and discussed with the panel, and valuable insight and feedback has been obtained. The progress of the PIRT panel can be followed by reviewing the following web site: http://www.nrc.gov/RES/pirt/BUC.

Planned Activities

Several planned activities that relate to the work in progress have been noted in the previous sections. Another planned activity scheduled to begin soon is a parametric study of the impact of control rod insertions on the SNF isotopic inventory and subsequent k_{eff} values in a cask environment. Also, work to review preshipment measurement approaches has been initiated by the NRC staff.

A major focus over the next year is to investigate various approaches for increasing the allowed inventory of SNF that can be inserted in a burnup-credit cask design. Using the current recommendations of ISG8, a significant portion of the current and anticipated SNF would not be allowed in a cask designed for burnup credit. If the restriction on burnable absorbers is removed, the potential inventory that can be considered for loading in a burnup credit cask will be expanded. However, the loading curves (burnup vs initial enrichment) developed with the current recommendations would be such that a large portion of the SNF inventory would be eliminated because the burnup value would be too low for the specified initial enrichment. Efforts in the coming year will seek to study various risk-informed approaches that may reduce the conservatism associated with development of the loading curve (i.e., lower the required burnup value needed for a specific initial enrichment). For example, the use of typical or average axial profiles may be acceptable if it can be demonstrated that the impact of using bounding profiles for a portion (some realistic upper limit based on the probability for multiple assemblies with atypical profiles) of the loading does not present an unacceptable risk to safety. Another example would be the assumption that BPRs are only used for one cycle even though the bounding case would provide for their use for three cycles. To investigate such approaches extensively will require additional information from industry regarding the range of parameter values (e.g., soluble boron concentration, moderator temperature) seen in typical and atypical reactor operations. Such information could allow the use of statistical analyses to help determine appropriate "typical" conditions and help assess the probability of "outlier" conditions that would normally be the basis for bounding values. *The goal is to develop criteria and/or recommendations that are technically credible, practical, and cost effective while maintaining needed safety margins*.

Of course a major component that will lower the loading-curve profile is the inclusion of fission products. The current work to estimate a "certified" fission product margin using the CERES experiments and the SCALE code system needs to be expanded to investigate more general approaches that might provide acceptable means for taking fission-product credit.

All of the work discussed in this paper has focused on PWR spent fuel. Subsequent to completion of that work, it is anticipated that similar efforts will be pursued to develop technical bases and guidance for BWR spent fuel.

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