Interim Information Report*: Technical Bases to Support Recommendations and Proposed Guidance for Expansion of ISG-8, Revision 1

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ABSTRACT

This report summarizes the technical bases to support recommendations and proposed guidance for expanded use of burnup credit in PWR transport and storage applications developed under the NRC/RES research program. The issuance of Interim Staff Guidance 8, revision 1 (ISG8-r1), has provided the impetus for industry to proceed with a new generation of high-capacity cask designs using burnup credit. However, adherence to the recommendations of ISG-8r1 will significantly limit the population of present and future spent nuclear fuel (SNF) that can reside in a burnup credit cask. Similarly, clarifying guidance on acceptable technical analysis approaches to address selected issues identified in ISG-8r1 should help expedite the licensing process for burnup credit cask designs. Four areas within ISG-8r1 have been identified as limitations to the practical usefulness of burnup credit and the NRC/RES program has initiated and performed research to provide technical bases to support recommendations and proposed guidance for near-term expansion of ISG-8r1 in each of the four areas. The research and resulting recommendations are summarized in this report. In addition, future research directions to pursue risk-informed approaches for increasing the allowed inventory of SNF that can be inserted into a high-capacity burnup credit cask are also briefly discussed.

1 INTRODUCTION

The issuance of Interim Staff Guidance 8, revision 1, (ISG-8r1)[1] has provided the impetus for industry to proceed with a new generation of high-capacity cask designs using burnup credit in the criticality safety evaluations. The first applications for PWR burnup credit casks are anticipated in early 2001. However, adherence to the recommendations of ISG-8r1 will significantly limit the population of SNF that can reside in a burnup credit cask (e.g., no fuel with burnable absorbers, insufficient credit due to conservatisms, etc.) and may restrict the use of high-capacity burnup credit casks for fuel with high initial enrichments (due to the corresponding burnup requirements). Similarly, clarifying guidance on acceptable technical analysis

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approaches to address selected issues identified in ISG-8r1 should help expedite the licensing process for burnup credit cask designs. The following four areas within ISG-8r1 have been identified as limitations to the practical usefulness of burnup credit:

1) limitation on cooling time to 5-years;
2) restriction on assemblies that have used burnable absorbers;
3) lack of guidance regarding acceptable approaches for treating the axial distribution of burnup; and
4) loading offset for enrichments greater than 4.0 wt% ^{235}U.

The NRC/RES program has initiated and performed research to provide technical bases to support recommendations and proposed guidance for near-term expansion of ISG-8r1 in each of the four areas. The research and resulting recommendations are briefly summarized in this report. Planned research directions (beyond closure of the aforementioned four technical areas) to pursue approaches for increasing the allowed inventory of SNF that can be inserted into a high-capacity burnup credit cask are also briefly discussed.

2 BACKGROUND

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 and without fixed burnable absorbers. 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 inherent 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 financial and safety-related 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 are considered. A related, significant complication lies in the desire for general storage and transport casks (for a given reactor type, PWR or BWR) to be qualified to accept SNF assemblies from all (or many) United States (U.S.) utilities, and thus accept fuel that has experienced potentially widely varying reactor operating conditions. In contrast to the fresh fuel assumption, the use of burnup credit requires validation of calculational methods used to predict the SNF nuclide compositions applied in the safety analyses. In addition, validation of analysis methods for the prediction of the neutron multiplication factor (k_{eff}) must include more nuclides and non-uniform isotopic distributions. 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 aided the development of approaches for a safety evaluation.

However, a consensus has not been reached on many important issues that affect the implementation of burnup credit, and subsequently the usefulness and associated benefits.

To date, there has been no regulatory experience in the U.S. with licensing of a PWR or BWR cask using burnup credit. U.S. industry and the U.S. Department of Energy (DOE) have supported a significant number of technical investigations (focused primarily on PWR fuel) to provide a foundation for implementation of burnup credit is this country. Based on these technical investigations, as well as others, the Spent Fuel Project Office (SFPO) of the Nuclear Regulatory Commission (NRC) issued Interim Staff Guidance 8 (ISG-8r0)[2] entitled *Limited Burnup Credit* in May 1999, which was the first U.S. regulatory guidance for using burnup credit. Supported by confirmatory research, Revision 1 (ISG-8r1)[1] to ISG-8r0 entitled *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks* was released in late July 1999. A discussion of the technical considerations that helped form the development of ISG-8r1 can be found in Ref. 3. Although subsequently incorporated within the NRC/SFPO draft standard review plan for transport, reference will made to ISG8-r1 throughout this report.

ISG-8r1 provides fairly specific recommendations for licensees and NRC staff in the preparation and review of criticality safety analyses for PWR casks using burnup credit and represents a significant expansion in the allowable use of burnup credit, in comparison to ISG-8r0. Some of the recommendations in ISG-8r1 limit the amount of burnup credit that can be utilized in the safety evaluation (e.g., credit for fission products is not included) and some limit the SNF population (type and range of characteristics) that would be allowed in a burnup credit cask. These recommendations were based on the technical information available to the NRC staff at the time, a desire for consistency with the industry standards developed for criticality safety of fissionable materials (ANSI/ANS –8.1) and LWR fuel (ANSI/ANS-8.17) in operations outside reactors, and with the recognition that experience and additional research would provide a basis for additional/expanded guidance.

The recommendations within ISG-8r1 limit the amount of burnup credit to that available from actinide compositions in SNF with an assembly-average burnup of 40 GWd/t or less and a cooling time of 5 years. In addition, the recommendations do not include burnup credit for assemblies that have used burnable absorbers. The ISG-8r1 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 included, but for each 0.1 wt% increase above 4.0 wt% ^{235}U, the assigned burnup loading value must be 1 GWd/t higher than the credited burnup used in the safety analysis. This loading offset was established based on engineering judgement to account for the lack of isotopic assay data for SNF with initial enrichments greater than 4.0 wt% ^{235}U. The ISG-8r1 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 variations in reactor operating histories, a lack of measured data for validation, and spatial variations of burnup within an assembly (axial and horizontal) should be accounted for in the safety analysis. Design-specific analyses to estimate the additional reactivity margin available from fission products and actinide nuclides not included in the licensing basis are recommended. The estimated margins should be assessed against estimates of uncertainties and potential non-

conservatisms. Furthermore, ISG-8r1 recommends the use of a burnup measurement prior to or during the loading procedure to ensure that each assembly is in compliance within the loading specifications for approved contents. The recommendations for a bounding approach and preshipment measurements are consistent with the international regulations[4] for the transport of fissile material.

As mentioned, the recommendations in ISG-8r1 were based on the technical information available to the NRC staff at the time and with the recognition that experience and additional research would provide a basis for additional/expanded guidance. To support the latter, NRC/RES initiated a general research program to address regulatory needs for safe, simple, and cost-effective implementation of burnup credit. The goal of the research program is to provide information that can serve as a basis for decisions on potential future modifications to the ISG-8 recommendations. Such future modifications should lead to enhanced usage of burnup-credit casks while maintaining an adequate margin of safety. The NRC is also 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 purpose of this report is to summarize the technical bases and proposed guidance for expanded use of burnup credit in PWR transport and storage applications developed under the NRC/RES research program. In addition, planned research directions (beyond closure of the aforementioned four technical areas) to pursue approaches for increasing the allowed inventory of SNF that can be inserted into a high-capacity burnup credit cask are briefly discussed.

3 RESEARCH AND RECOMMENDATIONS FOR NEAR-TERM EXPANSION OF ISG-8R1

Four areas within ISG-8r1 have been identified as limitations to the practical usefulness of burnup credit. Consequently, the NRC/RES research program has initiated efforts in each of the four areas to develop the necessary technical bases to support recommendations for near-term expansion of ISG-8r1. The existing ISG-8r1 limitations, research to address and/or resolve the limitations, and associated recommendations for each of the four areas are briefly described in the following sections.

3.1 SNF Cooling Time

SNF cooling time refers to the post-irradiation time period, beginning when the assembly is finally discharged from the reactor. ISG-8r1 recommends that the licensing basis be developed based on a single cooling time of 5 years. This recommendation (by itself) will not prevent cask loading of the vast majority of SNF assemblies currently in storage and simplifies the licensing and loading process by limiting the burnup credit loading curves (required burnup as a function of initial enrichment) to a single cooling time. Also, this approach circumvents the need to consider the initial peak reactivity immediately after discharge (within the first 100 hours). However, restricting the cooling time to a fixed value of 5 years eliminates assemblies with

shorter cooling times from cask loading and, more importantly, limits the allowable credit for reactivity reduction associated with longer cooling times.

3.1.1 Effect of Cooling Time on Reactivity

To assess the possibility of expanding the current cooling time guidance for dry cask storage and transport, detailed analyses[5] have been performed to demonstrate and quantify the reactivity behavior of SNF as a function of cooling time. The effect of cooling time on reactivity for various initial enrichments, burnups, and selected nuclide sets has been investigated. Further, the benefits of additional credit for cooling time have been quantified based on a realistic rail-type cask designed for burnup credit, as well as a four-assembly truck-type cask. The rail-type cask used for this and other studies throughout this report is the generic burnup credit (GBC-32) cask[6] shown in Figure 1. While the study primarily focused on dry storage and transport, analyses were extended out to 100,000 years to understand the relevant concerns associated with long-term disposal and their possible influence on dry storage and transport practice.

SNF discharged from a reactor will increase in reactivity for approximately 100 hours after discharge due to the decrease in neutron absorption caused by the decay of very short-lived fission products. The decrease in reactivity from 100 hours to 100 years is driven by the decay of the ^{241}Pu fissile nuclide ($t_{1/2}$ = 14.4 years) and the buildup of the neutron absorbers ^{241}Am (from decay of ^{241}Pu) and ^{155}Gd (from ^{155}Eu which decays with $t_{1/2}$ = 4.7 years). After about 50 years the ^{155}Gd buildup is complete and the ^{241}Pu has decayed out by approximately 100 years. After this time the reactivity begins to increase, governed primarily by the decay of two major neutron absorbers – ^{241}Am ($t_{1/2}$ = 432.7 years) and ^{240}Pu ($t_{1/2}$ = 6,560 years) – and mitigated somewhat by a decrease in the fissile inventory as ^{239}Pu ($t_{1/2}$ = 24,100 years) decays and causes an increase in ^{235}U. After approximately 30,000 years, the ^{240}Pu and ^{241}Am decay is complete and the reactivity again begins to decrease as the decay of ^{239}Pu dominates the process. The behavior is illustrated in Figure 2.

3.1.2 Discussion

The location of the SNF reactivity minimums and maximums and their values are dependent upon the nuclides included in the reactivity calculations. Also, the extent of the reactivity minimums and maximums has been shown[5] to be dependent on the assembly initial enrichment and discharge burnup. Figure 2 shows an example of the reactivity behavior as a function of cooling time (with the axial burnup distribution included) in the GBC-32 cask for the various nuclide sets defined in Table 1. Based on the behavior shown, it is not straightforward to select a practical, yet bounding value for the cooling time in a safety evaluation that includes burnup credit. Using Figure 2 as representative, it can be observed that the fixed 5-year cooling time of ISG-8r1 provides a conservative reactivity relative to that of longer cooling times (up to several thousand years for actinide-only burnup credit). However, the 5-year cooling time is not bounding relative to the reactivity of shorter cooling times. Thus, as Figure 2 indicates, it appears best to select a time frame of interest for the application and develop criteria that provide for implementation within the given time frame.

Although the time frame of interest to dry storage and transport is not well defined, it is assumed to be between 1 and 200 years. Storage and transport systems are currently licensed for periods of 20 years, and thus, 200 years represents 10 license periods (i.e., the initial license followed by

9 license renewals). Considering typical plant-life expectations (on the order of 40 years) and that SNF may reside in a spent fuel pool for a short period of time after final reactor shutdown (probably less than 10 years), it is expected that fuel loaded into dry storage and transport casks will have cooling times less than 50 years. Therefore, credit for cooling time beyond 50 years does not seem to be of any direct benefit for current storage and transport analyses. Furthermore, Figures 2 and 3 indicate that there is insignificant reactivity credit to be gained from consideration of cooling times beyond 50 years. Referring back to the 200-year time frame assumed for dry cask storage and transport, the results in Figure 2 indicate that burnup credit for cooling times out to 50 years can potentially present some long-term concerns if the SNF remains in dry storage long enough for the reactivity to rise above the reactivity associated with the cooling time used in the safety analysis. However, the results also indicate that for all three classes of burnup credit the reactivity remains below the 40-year level for all cooling times greater than 40 years and less than 200 years. A limit of 40 years also corresponds well to the 50-year maximum cooling time proposed above based on practical benefits to dry storage and transport. A similar argument (with added conservatism) led to the maximum cooling time limit of 25 years proposed in the DOE topical.[7]

Assuming a practical lifetime of 200 years for dry storage, the technical information discussed above indicates that the cooling time to use in burnup credit evaluations should be between 1 and 40 years. An important point to note is that the uncertainty associated with reactivity changes due to cooling time in the 1-to-40-year time period should be very small because decay data important to changes in this time period are known with very good accuracy.[8]

Administrative procedures that ensure confirmation of cooling time during cask loading and proper use of loading curves will need to be developed for use with burnup credit. If the ISG-8r1 recommendations are expanded to allow multiple cooling times, each cask license may contain a separate loading curve for each cooling time of interest. Current loading procedures have limits on the following parameters: initial enrichment, burnup and cooling time (for shielding and decay heat considerations), and assembly design. In addition to the current limits, loading procedures for burnup credit casks will necessarily include loading (burnup-enrichment) curves that may be assembly dependent and procedures for verification of accumulated burnup. The increase in the number of loading curves due to a use of multiple cooling times will result in increased complexity in the cask loading procedures and corresponding complication in the administrative controls. To provide some bound to this increased complexity, it may be prudent to place a limit on the total number of cooling times considered.

As indicated initially in this section, the limit on cooling time allowance hinges on the time frame of interest. Should it be considered plausible that some unanticipated scenario might cause SNF to remain in a dry storage cask beyond the 200 years assumed here, then the cooling time allowance may need to be reconsidered based on the amount of burnup credit allowed. For actinide-only burnup credit, Figure 2 shows that the absolute minimum reactivity as a function of time is always greater than the absolute maximum reactivity based on best-estimate calculations (i.e., all nuclides present). Thus, storage casks licensed with actinide-only assumptions (consistent with the ISG-8r1 recommendations) would have sufficient subcritical margin to accommodate storage beyond the 200-year time frame. However, if future cask licensing includes credit for fission products, this subcritical margin is no longer available and one must

consider whether the risk of storage beyond 200 years warrants limiting the cooling time credit to a value less than 40 years. In the event that it does, it has been suggested that a value of 10 years be assumed as the cooling time limit for safety analyses. The rationale for this limit is that, except for SNF that is highly under-burned (e.g., 5.0 wt% ^{235}U, 20 GWd/t), the best-estimate results (i.e., all nuclides present) for k_{eff} at a 10-year cooling time are always greater than the maximum k_{eff} at the secondary peak (10,000-to-30,000-year time frame).

A final issue for discussion is the potential sensitivity of the cooling time allowance to the axial burnup distribution. The axial burnup distribution decreases the reactivity reduction associated with cooling time, in comparison to that anticipated with a uniform axial burnup distribution. Thus, emphasizing the need for proper treatment of the axial burnup distribution. Initial studies[5] have confirmed that bounding axial burnup profiles are fairly insensitive to cooling time, hence a bounding axial burnup profile for one cooling time is expected to be bounding for all cooling times within the 1-to-40 year time frame.

3.1.3 Recommendations

For burnup credit criticality safety analyses performed at 5 years, increasing the cooling time of the SNF results in an increasingly conservative safety margin out to approximately 100 years. The magnitude of the conservatism depends on the initial enrichment and burnup of the fuel. Additional conservatism may be added by basing calculated isotopic compositions on a shorter assumed cooling time period, i.e., cooling periods as short as one year. However, there is no apparent justification for the additional conservatism and the reduction in reactivity associated with cooling time in the range of 1 to 100 years is well established. Therefore, expansion of the cooling time considered in ISG8r1 is recommended.

The analyses and discussions summarized in this report and described in greater detail in Ref. 5, provide a technical basis for revising ISG-8r1 to allow burnup credit for cooling times between 1 and 40 years. Additionally, the analyses show that expansion of credit for cooling times outside of the 1-to-40 year range does not yield significant benefits. Finally, it is noted that these recommendations are made for actinide-only burnup credit, assuming a practical cask lifetime of approximately 200 years. Consequently, future expansions to allow credit for fission products may necessitate a reduction in the upper bound of the allowed cooling time range (e.g., reduction from 40 years to 10 years).

3.2 Burnable Absorbers

The ISG-8r1 recommendations do not include burnup credit for assemblies that have used burnable absorbers. This limitation eliminates a large portion of the currently discharged spent fuel assemblies from cask loading, and thus severely limits the practical usefulness of the burnup credit guidance. Burnable absorbers may be classified into two distinct categories: (1) burnable poison rods (BPRs) and (2) integral burnable absorbers (IBAs). 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, IBAs refer to burnable poisons that are a non-removable 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. Because of the differences in the way BPRs and IBAs are used, and the corresponding differences in their impact on the reactivity of SNF, they are discussed separately below.

3.2.1 Burnable Poison Rods (BPRs)

Several different BPR designs have been used in commercial nuclear reactors. However, all BPR designs are similar in that they contain thermal neutron absorbing material in rods sized to fit within the assembly guide tubes. Burnable poison rod assemblies (BPRAs) are typically inserted into a PWR fuel assembly during its first cycle in the reactor core and, depending on the design, the actual number of BPRs within a BPRA or the poison loading in the BPRs are variable. BPRAs are typically used in an assembly during its first cycle to suppress the initial excess reactivity.

To assess the possibility of relaxing the current restriction on BPRs for dry cask storage and transport, detailed analyses[9] have been performed to demonstrate and quantify the reactivity behavior of SNF for various BPR designs, exposure conditions, initial enrichments, and nuclide sets. All BPR types that have been widely used in U.S. PWR reactors were considered, including Westinghouse BAAs, Westinghouse WABAs, and B&W BPRs. Detailed BPR specifications are available in Ref. 9.

3.2.1.1 Effect of BPRs on Reactivity

The presence of BPRs during depletion hardens the neutron spectrum because of the removal of thermal neutrons by capture in ^{10}B and, more importantly, 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.

To investigate the effect of BPRs on reactivity, depletion calculations were performed assuming BPRs 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. The 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 4 plots the Δk values (relative to no-BPR depletion calculations) as a function of burnup using the major actinides from Table 1 (i.e., set 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 Figure 4 demonstrate that the reactivity difference 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 Figure 4, but with the inclusion of the major fission products, the reactivity behavior is very similar to that of the actinide-only condition.

For comparison of the effect of the various BPR designs, calculations have also been performed[9] for the Westinghouse BAAs and the B&W BPRs. The results are very similar to those shown in Fig. 4 for the Westinghouse WABAs. Since the B_4C weight percent is known to vary in the B&W BPRs, the reactivity effect of varying the B_4C poison loading was also investigated and verified to increase with poison loading. Finally, the reactivity effect of BPRs was found to increase with decreasing initial enrichment (for a fixed burnup).

Analysis of the GBC-32 cask provides k_{eff} values for actinide-only and actinide-plus-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.

3.2.1.2 Discussion

The reactivity difference due to BPRs increases nearly linearly with burnup and is dependent upon the number and poison loading of the rods and the initial fuel enrichment. For a fixed burnup, the reactivity difference due to BPRs increases with decreasing initial enrichment and with increasing poison loading (either number of BPR rods or ^{10}B wt%). Although variations are observed for the different BPR designs, maximum reactivity increases have been found to be ~1 to 3% Δk when maximum BPR loading and exposure time are assumed for typical initial enrichment and discharge burnup combinations. Expected typical reactivity increases, based on one-cycle exposure, were found to be less than 1% Δk. Of the BPR designs considered, the Westinghouse BAA design yielded the greatest positive reactivity effect. Although BPR poisons are effectively depleted during the first cycle of exposure, the reactivity difference is primarily due to the displacement of moderator.

While it is known that BPRs are typically inserted into an assembly during its first cycle of operation and subsequently withdrawn and discarded, this practice is not exclusive. Provided analyses were accompanied by administrative restrictions to ensure that assemblies with greater than one cycle of BPR exposure were not acceptable for loading, analyses could be performed based on only a single cycle of BPR exposure. However, such an approach would require the maximum single cycle exposure to be defined to assure all single cycle BPR exposures are bounded. A complication associated with this approach is the necessity of plant data specifying assembly BPR exposure. Considering the magnitude of the conservatism associated with assuming the BPRs are present throughout the entire irradiation, the additional complexities of such administrative controls may be acceptable. Alternatively, it may be possible to credit the

conservatism associated with the maximum exposure approach to account for the effects of temporary control rod insertions. BPRs and control rods both cause localized spectral hardening, and thus, result in similar reactivity effects. Because it is physically not possible for a BPRA and a control rod assembly to be inserted into a fuel assembly at the same time, the BPR modeling assumption may be used to bound realistic combinations of BPR and control rod exposures.

3.2.1.3 Recommendations

The analyses and discussions summarized in this report and described in greater detail in Ref. 9, provide a technical basis for removing the ISG-8r1 restriction on assemblies exposed to BPRs. Guidance should require analyses to include the effect of BPRs for assemblies that are classified as acceptable contents for the particular cask. For example, safety analyses for casks that are to be loaded with assemblies that contained BPRs during irradiation should account for the limiting realistic BPR irradiation justified by the applicant's operations and design information and/or verified during cask loading. 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. However, more realistic approaches based on typical operating conditions and/or loading restriction are acceptable with supporting justification (e.g., loading verification, analyses of statistically representative plant operating data, consideration of the impact on reactivity associated with loading assemblies that have greater than assumed BPR exposure, etc.).

3.2.2 Integral Burnable Absorbers (IBAs)

Numerous Integral Burnable Absorber (IBA) types have been used in commercial nuclear fuel assembly designs to suppress the initial reactivity. Variations in the IBA poison material, composition, placement within rods, and rod configurations exist among current PWR IBA fuel assembly designs. These IBA characteristics are varied in combination with the initial fuel assembly enrichment and core location to achieve core operating and fuel management goals. To assess the possibility of relaxing the current restriction on IBAs for dry cask storage and transport, detailed analyses[10] have been performed to demonstrate and quantify the reactivity behavior of SNF for various IBA types/designs, conditions, and initial enrichments. These IBA types include IFBA rods, UO_2-Gd_2O_3 rods, UO_2-Er_2O_3 rods, and Al_2O_3-B_4C rods. Detailed IBA specifications are available in Ref. 10. Based on the available data, analyses were performed for a representative, realistic range of fuel initial enrichment and poison loading combinations representative of actual assemblies. All IBA types that have been widely used in U.S. PWRs were included in the evaluation.

3.2.2.1 Effect of IBAs on Reactivity

For PWR fuels without IBAs, the reactivity decreases with burnup in a nearly linear fashion. In contrast, for PWR fuel assembly designs that make significant use of IBAs, the reactivity actually increases as fuel burnup proceeds, reaches a maximum at a burnup where the IBA is nearly depleted, and then decreases with burnup in a nearly linear fashion. For fuel assembly designs that make modest use of IBAs, the reactivity decreases with burnup slowly up to the point where the IBA is nearly depleted, and then decreases with burnup in the nearly linear manner. The assemblies are typically designed such that the burnable absorber is effectively

depleted in the first third of the assembly life, and as a result, the assembly reactivity typically peaks within this period of burnup. The reactivity behavior of a PWR fuel assembly with and without IBAs (neutron poisons) present as a function of burnup is illustrated in Figure 5.

The presence of IBAs during depletion hardens the neutron spectrum, 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 can potentially increase the reactivity of the fuel at discharge and beyond, depending on the IBA assembly design characteristics. However, as mentioned, the assemblies are typically designed such that the burnable absorber is effectively depleted in the first third of the assembly life, and thus is exposed to a hardened spectrum during the first third of its exposure only. Note that, unlike BPRs, which are inserted into assembly guide tubes, IBAs do no displace moderator in the assembly lattice, and thus have a less significant impact on the neutron spectrum.

The detailed analyses presented in Ref. 10 demonstrate that the neutron multiplication factor for an assembly without IBAs is always greater (as a function of burnup) than the neutron multiplication factor for an assembly that utilized any of the following IBA types: UO_2-Gd_2O_3, UO_2-Er_2O_3, or Al_2O_3-B_4C rods. Conversely, the neutron multiplication factor for an assembly with IFBA rods present was found to exceed (typical values between 0.1 and 0.3 % Δk, maximum of 0.4% Δk) the neutron multiplication factor for an assembly without IFBA rods. Therefore, neglecting the IBAs in a burnup-credit criticality safety analysis will yield conservative results for assembly designs with UO_2-Gd_2O_3, UO_2-Er_2O_3, or Al_2O_3-B_4C IBA rods and non-conservative results for assembly designs with IFBA rods. In all cases, for burnups characteristic of discharge, the reactivity effect of IBAs is relatively small (less than ~1.0% Δk) and generally well behaved.

3.2.2.2 Discussion

These analyses demonstrate that assembly designs with UO_2-Gd_2O_3, UO_2-Er_2O_3, or Al_2O_3-B_4C IBA rods are less reactive throughout burnup than their corresponding designs without the IBA rods (i.e., non-poisoned, equivalent enrichment), and thus provide justification for neglecting the presence of these IBA designs in a burnup credit criticality safety evaluations. However, for assembly designs with IFBA rods, the positive reactivity effect must be appropriately addressed.

Although the analyses do not address the issue of validation of depletion methods for assembly designs with IBAs, they do demonstrate that the effect of the IBAs is relatively small (at or near target discharge burnups) and generally well behaved. The results follow physics-based expectations and are in close agreement with a similar, albeit less detailed, study[11] that was performed with an independent 2-D core physics code. Furthermore, the recommended approaches for addressing fuel assemblies with IBAs, as described below, do not involve explicit analyses with IBAs present, and thus do not necessitate validation of the depletion methods for assembly designs with IBAs.

3.2.2.3 Recommendations

The analyses summarized in this report and described in greater detail in Ref. 10, provide a technical basis for revising ISG-8r1 to include burnup credit for assembly designs with Integral Burnable Absorbers (IBAs). With the notable exception of the Westinghouse IFBA rods, the

neutron multiplication factor for an assembly without IBAs is always greater (throughout burnup) than the neutron multiplication factor for an assembly with IBAs, including UO_2-Gd_2O_3, UO_2-Er_2O_3, and Al_2O_3-B_4C rods. Therefore, for those IBAs other than IFBAs, burnup credit criticality safety analyses may simply and conservatively neglect the presence of the IBAs by assuming non-poisoned, equivalent enrichment fuel. Considering the variations in IBA assembly designs, neglecting the presence of the IBAs is an important simplifying assumption that does not add significant unnecessary conservatism.

For assembly designs with IFBA rods, the neutron multiplication factor was found to be slightly greater (maximum of 0.4% Δk) than the neutron multiplication factor for an assembly without IFBA rods. Therefore, the positive reactivity effect due to the presence of IFBA rods should be considered in any burnup-credit criticality safety analysis seeking to qualify IFBA assemblies as acceptable contents. Due to the significant variations in IFBA assembly designs, simple strategies for addressing the positive reactivity effect are desirable. Two possible strategies for consideration include: (1) the inclusion of a small reactivity bias to bound the effect of the IFBA rods, or (2) demonstration that the effect of the IFBA rods is bounded by analysis that accounts for the effect of BPRs. While feasible, the use of a small reactivity bias would require justification for the value of the bias. The use of a constant, uniform bias that bounds all cases would result in conservatism with respect to "typical" cases, but the conservatism would be very small. Alternatively, it will be simpler and less burdensome to demonstrate that the effect of the IFBA rods is bounded by the BPR modeling approach. Comparison of the reactivity effect of IFBA rods (shown in Ref. 10) to the reactivity effect of BPRs (as quantified in Ref. 9), clearly demonstrates that the reactivity effect of the IFBA rods is significantly less than the reactivity effect due to BPRs. Furthermore, considering the fact that BPRs are seldom used within assemblies that have IFBA rods, and when used, are employed in a limited way (e.g., a small number of BPRs may be used in conjunction with an assembly that has a relatively light IFBA loading), reliance on the BPR modeling to account for the effect of IFBA rods is justified. However, this approach would only be applicable to analyses that consider BPR exposure.

3.3 Axial Burnup Distribution

ISG-8r1 recommends the use of analyses that provide an "adequate representation of the physics" and notes particular concern with the axial and horizontal variation of burnup. The horizontal variation of burnup has been investigated within the context of the development of Ref. 7. Due to the relatively minor impact of the horizontal burnup profile on the neutron multiplication in a typical rail-type burnup credit cask,[5] further investigations related to horizontal burnup profiles currently have low-priority, but may be considered in future work. In contrast, the axial burnup profile has a significant impact on reactivity, and therefore is an extremely important component of a burnup credit safety analysis. However, ISG-8r1 offers no guidance regarding an acceptable means to address the axial burnup distribution. Consequently, some potential applicants have expressed that better guidance for an acceptable approach to address the axial burnup distribution would expedite their submittal of a burnup credit application.

3.3.1 Effect of Axial Burnup Distribution on Reactivity

Axial variations in flux, which are mainly due to leakage at the fuel ends, result in a non-uniform burnup distribution along the axial length of the fuel. The axial distribution is characterized by end regions that are significantly under-burned with respect to the assembly-average burnup. The shape of the distribution is dependent upon the accumulated burnup, as well as other characteristics of the assembly operating history. For fuels of moderate-to-high burnup (i.e., burnups beyond approximately 20 to 30 GWd/t), these under-burned regions are dominant in terms of reactivity, and thus, must be properly represented to ensure subcritical margins.

Numerous studies have been performed to quantify the reactivity effect associated with axial burnup distributions. A fairly comprehensive review of those studies is available in Ref. 12. In general, these studies have shown that assuming uniform axial burnup is conservative for low burnups, but becomes increasingly nonconservative as burnup increases. The transition between conservative and nonconservative is dependent on numerous factors, but generally occurs in the burnup range of 20 to 30 GWd/t. These studies concluded that for fuels of moderate-to-high burnup, the reactivity difference between analyses with explicit representation of the axial burnup distribution and analyses that assume uniform axial burnup is positive. The amount by which the axial burnup distribution increases reactivity has been shown to be dependent upon many factors, but is primarily dependent on the slope of the burnup profile near the ends of the fuel. The profile is dependent on the fuel assembly design, burnup, and the operating conditions of the reactor. Recognizing the importance of the axial burnup profile, work sponsored by the DOE has provided a database[13] of more than 3000 PWR axial-burnup profiles, and studies[14] 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 represents typical and atypical profiles resulting from irradiation in U.S. PWR reactors.

3.3.2 Discussion

As mentioned, ISG-8r1 is ambiguous on how to properly account for the axial burnup distribution; particularly the selection of axial burnup profiles for the safety analyses. The existing axial burnup profile database prepared by Yankee Atomic (Ref. 13) contains 3169 PWR axial burnup profiles, which represent three fuel vendors through the mid-1990s, 20 different reactors, and 106 cycles of operation. The 106 cycles of operation include first cycles, out-in fuel management and low leakage fuel management.[13] The axial burnup profiles in the database were calculated with various three-dimensional core physics codes. These codes are the current methods used in fuel management, reload analysis, and core operational support by the nuclear industry, and thus their accuracy is verified through safe reactor operation and industry history with operating within technical specifications. However, there has been a great deal of interest in quantifying the uncertainties in calculated burnups, particularly as a function of axial height. Responding to this interest, a study[15] was performed to evaluate the uncertainties in the burnup of fuel assemblies utilizing in-core measurements and core neutronic calculations for a Westinghouse PWR. The study[15] concluded that the uncertainty in burnup, evaluated over three cycles of operation, decreases with increasing burnup. For assemblies discharged after one cycle of burnup the uncertainty was estimated to be 1.90%; after two cycles of burnup, the uncertainty is 0.98%; and after three cycles of burnup, the uncertainty is 1.02%. The decrease in uncertainty after two-cycles of burnup is attributed[15] to the self-correcting nature of burnup. The part of the

study that is particularly relevant to this discussion is the evaluation of the uncertainty in the axial distribution. Uncertainties of less than 7% are quoted[15] for the top and bottom ends of the assemblies.

The breakdown of the 3169 profiles, in terms of fuel vendor/design, are as follows: 1334 B&W 15x15 profiles, 544 Combustion Engineering (CE) 14x14 profiles, 228 CE 16x16 profiles, 156 Westinghouse 15x15 profiles, and 907 Westinghouse 17x17 profiles. The data covers a range of burnup from 3.086 to 55.289 GWd/t and an enrichment range of 1.24 to 4.75 wt% ^{235}U. To illustrate the range and depth of the database, in terms of burnup and enrichment, the profiles have been divided into burnup and enrichment groups, where each group spans a burnup interval of 5 GWd/t and an enrichment interval of 0.5wt % ^{235}U. The detailed breakdown of the number of profiles that fall within the burnup and enrichment groups is provided in Table 2. The profiles include fuel designs that used burnable absorbers with different poison absorber types such as: Burnable Poison Rods (BPRs) of borosilicate glass and B_4C; and Integral Burnable Absorbers (IBAs) of ZrB_2 (Integral Fuel Burnable Absorbers, IFBAs), B_4C, erbium and gadolinium. In addition, the profiles include assemblies exposed to control rods, including Axial Power Shaping Rods (APSRs). Thus, in terms of categories, the axial profile database provides an excellent representation of discharged PWR SNF assembly designs.

Although the database is comprehensive, it is not exhaustive. One of the expressed concerns has been the adequacy of this (or any) finite database to completely represent the infinite variety of possible profiles resulting from irradiation in U.S. PWRs. To address this concern, a statistical evaluation[16] has been performed on the neutron multiplication factors resulting from the profiles contained in the database to assess the likelihood of the existence of significantly more reactive axial burnup profiles and the associated consequence to the neutron multiplication. Based on the available database, the reactivity effect of each of the axial burnup profiles was calculated and the mean and standard deviation for each of 12 burnup ranges (as defined in Ref. 14) was determined. The results have been used to (1) assess how representative the most limiting profiles are to the rest of the profiles and (2) provide an indication of the probability that other axial profiles may exist that are more reactive than the limiting profiles (from the database).

To enable the statistical examination, a criticality calculation was performed for each of the profiles in the database. The calculations have generally confirmed the bounding profiles determined in Ref. 14, but have shown that the bounding profiles are not representative of the average. Figure 6 shows the spread of k_{eff} values that result from the set of profiles in one of the 12 burnup ranges considered, together with the bounding "real" (i.e., actual profile from the database) and "artificial" (as defined in Ref. 14) profiles. Similar figures have also been generated[16] for the other burnup ranges, but are not include here for brevity. In addition to the individual calculated k_{eff} values, Figure 6 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 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 representative of typical SNF profiles. Consequently, one can infer that the probability that other axial profiles exist that are notably more reactive than the limiting profile (determined from the database) is very small. 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 results in significant conservatism in comparison to reality.

Building on the results of the statistical examination, analyses were also performed to assess the impact of loading an assembly into a burnup credit cask that has an axial profile that is not bounded by the exiting database. The analysis has confirmed that the reactivity consequence is not significant (less than ~0.2% Δk, but depends on burnup and the specification of the "more bounding" profile). The "more bounding" profiles used for the analysis were artificially developed profiles created by setting the average node burnup equal to the minimum node burnup for the actual bounding profiles.[16] Considering that (1) the limiting profiles are based on statistical outliers, (2) the limiting profiles will be applied to all assemblies in a burnup credit cask, (3) the very small probability that more reactive profiles exist, and (4) the small reactivity consequence of loading an assembly with an artificially-developed "more bounding" profile, the use of limiting profiles from the Yankee Atomic database should provide significant conservatism (~1-2% Δk, depending on burnup) in comparison to reality.

As evidenced from Figure 6, 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 a more realistic 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 also being investigated.

3.3.3 Recommendations

The analyses summarized in this report and described in greater detail in Ref. 16, provide a technical basis for revising ISG-8r1 to endorse the adequacy of the existing Yankee Atomic axial burnup profile database for obtaining profiles for use with actinide-only burnup credit within the currently established burnup and enrichment range (i.e., ≤ 40 GWd/t and ≤ 4.0 wt% ^{235}U). The rationale for this recommendation are: 1) the Yankee Atomic axial profile database provides an adequate representation of discharged U.S. PWR SNF assembly designs; 2) the limiting profiles, as determined from the database, are statistical outliers, and thus the probability that more reactive profiles exist is very small; 3) the limiting profiles will be applied to all assemblies in a burnup credit cask; and 4) the small reactivity consequence of loading an assembly with an artificially-developed "more bounding" profile. Although there are more than 200 profiles for enrichments greater than 4.0 wt% ^{235}U and more than 350 profiles for burnups greater than 40 GWd/t, the recommendation is limited to the specified burnup and enrichment range defined in ISG-8r1 at this time. Future work should address the adequacy of the database for higher burnups and enrichments, and expand the database as additional profiles become available. Additionally, future work should also evaluate the impact of the burnup uncertainty.[15]

Applicants seeking to utilize more realistic profiles will need to provide measurement verification of the assembly burnup profile selected for cask loading and/or present justification based on risk-informed concepts. Applicants should be encouraged to expand the existing database.

3.4 Loading Offset

ISG-8r1 recommends a loading offset equal to at least 1 GWd/t for every 0.1 wt% ^{235}U increase above 4 wt% ^{235}U . The maximum enrichment that may be considered in any case is 5.0 wt% ^{235}U. Therefore, spent fuel with an initial enrichment of 4.5 wt% ^{235}U would be assigned a burnup loading value that is at least 5 GWd/t higher than the burnup credited in the safety analysis. The loading offset provides an additional conservative margin that accounts for potentially higher uncertainties in calculated isotopic inventories of SNF with initial enrichments greater than 4.0 wt% ^{235}U, a regime where there is currently no experimental data available to validate computer code isotopic predictions.

The loading offset has been identified as a potential limitation to the practical usefulness of burnup credit. In response, efforts have been expended to assess the impact of the loading offset and explore technical approaches aimed at removal of the loading offset. However, it should be noted that, for loading in a burnup credit cask, minimum burnup requirements increase with initial fuel enrichments. Consequently, considering the current ISG-8r1 limit on burnup (40 GWd/t), the benefits associated with removing the enrichment loading offset (without removing the limit on burnup) are not apparent for high-capacity, rail-type burnup credit casks. On the other hand, the enrichment loading offset does provide a means to store fuel up to 5.0 wt% ^{235}U enrichment with burnup credit in a currently licensed cask design (e.g., a 24-assembly cask) or in a four-assembly truck-type cask. Therefore, as stated, until the limit on burnup is either raised or removed, there does not appear to be any realizable, practical benefits associated with removing the loading offset. Furthermore, a review of currently discharged SNF reveals that less than 15% of the discharged SNF assemblies have initial enrichments greater than 4.0 wt% ^{235}U.

3.4.1 Rationale for Loading Offset

The ANSI/ANS-8.1 standard[17] outlines the requirements for code and data validation for nuclear criticality safety outside of reactors. The standard requires that calculational methods be validated by direct comparison with experimental results to establish the bias. The area of applicability may be extended beyond the range of experimental data and conditions by making use of trends in the bias and uncertainty. However, the standard states that “where the extension is large, the method should be supplemented by other calculational methods to provide a better estimate of the bias, and especially of its uncertainty in the extended area (or areas), and to demonstrate consistency of computed results.”

The present experimental database of public domain actinide assay data[18,19] consists largely of fuel with enrichments below 3.5 wt% ^{235}U, with only one measurement for fuel above 3.4 wt% ^{235}U (a 3.89 wt% ^{235}U sample with a low burnup of 12 GWd/t). The enrichment and burnup range of the spent fuel samples used in recent validation studies is shown in Figure 7. The figure illustrates the paucity of experimental data in the high-enrichment and high-burnup regimes. The loading offset provides a means of extending the usefulness of ISG-8r1 to include spent fuel with initial enrichments above 4 wt% ^{235}U, using an engineering approach to compensate for potentially larger uncertainties. Note that the ISG-8r1 limit of 4 wt% ^{235}U already extends beyond the available measured data.

3.4.2 Effect of Loading Offset

The loading offset, expressed in terms of Δk, is illustrated in Figure 8 for the GBC-32 cask design employing actinide-only burnup credit for a 5-year cooling time.[19] The reactivity margin for 5 wt% ^{235}U fuel, the maximum enrichment considered by ISG-8r1, ranges from approximately 0.035 to 0.045 Δk, depending on the fuel burnup. This added margin, shown in Figure 8, can be compared with the actinide isotopic uncertainties for which it is intended to compensate as a means of estimating the conservatism in ISG-8r1 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-V cross-section data.[20] 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 reflects the variance of the predicted bias and the number of assay measurements available.

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 Monte Carlo[21] or root-mean-square approach. However, the random method does not consider potentially correlated uncertainties in transmutation or decay chains. The actual effect is likely somewhere between these two approaches. Assuming the more conservative strategy, the net reactivity margin associated with the actinide uncertainties is illustrated in Figure 9 for a range of enrichments and burnup.[19] 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 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 random approach, the margin is reduced 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 9 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 ISG-8r1 loading offset above 4 wt% (see Figure 9) amounts to about an added 4% $\Delta k/k$ (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 ISG-8r1 loading offset (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.

3.4.3 Discussion

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.[22] 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[23] 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.

Several studies suggest that the effect of enrichment on the isotopic uncertainties should be minimal. The published French results[23] 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.[22] 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 4.0 wt% to 5.0 wt%). However, there is currently insufficient experimental data to definitively 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.

Another point of interest to this discussion is the requirement within ISG-8r1 for design-specific analyses to estimate the additional reactivity margin available from fission products and actinide nuclides not included in the licensing basis. It is suggested in ISG-8r1 that the estimated margins be assessed against estimates of uncertainties and potential nonconservatisms, including "any potential nonconservatisms in the models for calculating the licensing basis inventories." Examination of the estimated reactivity margins[6] available from fission products and minor actinides in the GBC-32 cask provides assurance that the negative reactivity margin associated with fission products and minor actinides will more than offset any additional isotopic uncertainties associated with the relatively minor increase in enrichment (i.e., from 4.0 wt% to 5.0 wt%).

3.4.4 Recommendations

There is a significant body of information that would support the elimination of the loading offset. The rationale for such a recommendation are: 1) preliminary sensitivity analyses indicate

that the data sensitivities change very little with increasing enrichment (within this range), suggesting that the existing isotopic data will be applicable for validation within this extended enrichment range, 2) published results by the French that indicate no significant trends for higher enrichment fuel, and 3) the negative reactivity margin associated with fission products will more than offset any additional isotopic uncertainties associated with this increase in enrichment. It should be noted that concomitant with increasing fuel enrichments are increasing complexities in fuel assembly designs. Therefore, the potential for larger isotopic uncertainties due to assembly design complexities should be investigated to assess the need for additional data.

A potential impediment to the elimination of the loading offset is the lack of experimental data available in this regime to support the recommendation, and consequently the difficulty in defending such a position based on a strict interpretation of the validation requirements of ANS-8.1. However, limited isotopic data will soon (anticipated Feb. 2001) be available in the range of 4.0-4.65 wt% ^{235}U to help quantify the uncertainties and substantiate the above recommendation.

4 RESEARCH DIRECTIONS FOR FUTURE EXPANSION OF ISG-8

Although the issuance of ISG-8r1 has provided the impetus for industry to proceed with design and licensing of a new generation of high-capacity cask designs using burnup credit, significant expansions in the allowable burnup credit will be necessary to accommodate nearly all of the SNF assemblies within burnup credit casks. In particular, adherence to the recommendations of ISG-8r1 will significantly limit the population of SNF that can reside in a burnup credit cask (e.g., no fuel with burnable absorbers, insufficient credit due to conservatisms, etc.) and may restrict the use of high-capacity burnup credit casks for fuel with high initial enrichments (due to the corresponding burnup requirements). The goal of the work described in the previous section was to provide technical bases and corresponding recommendations to eliminate identified restrictions on assemblies (e.g., assemblies that have used burnable absorbers) and increasing the allowable credit (e.g., additional credit for cooling time).

Although this work represents an important step in expanding the use of burnup credit to accommodate the majority of PWR SNF assemblies within burnup credit casks, a good deal more work is necessary. Several planned activities that relate to the work in progress have been noted in the previous section. Another activity that has been recently initiated 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. However, the major focus over the next year is to investigate approaches for increasing the allowed inventory of SNF that can be inserted in a burnup-credit cask design (i.e., pursuing approaches to significantly expand the allowable burnup credit).

As mentioned, using the current recommendations of ISG-8r1, a significant portion of the current and anticipated SNF would not be allowed in a cask designed for burnup credit. Obviously, adopting the recommendations described in the previous section will significantly expand the potential allowable inventory. As a result, a significant portion of the currently discharged SNF (more than 50%) could be loaded into a burnup credit cask. However, the loading curves (burnup vs initial enrichment) developed with the current recommendations are still such that a

notable portion of the SNF inventory would be eliminated because the burnup value would be too low for the specified initial enrichment. This point is illustrated in Figure 10, which shows an illustrative burnup-enrichment loading curve (based on current guidance) in comparison to discharged PWR SNF (through the end of 1998). One can see from Figure 10 that sufficient acceptable discharged SNF exists in current storage to motivate design and licensing of burnup credit casks. Further, it is apparent that even minor reductions in the minimum required burnup, for a given enrichment, will significantly increase the number of assemblies that may be accommodated. In other words, even minor reductions in analysis conservatism or increases in allowable burnup credit will lower the burnup-enrichment loading curve, and thus notably expand the usefulness of burnup credit. For example, an additional 3% Δk worth of burnup credit can lower the burnup-enrichment loading curve by 4-8 GWd/t (depending on the initial enrichment), which would significantly increase the number of assemblies that would be acceptable for loading (see Figure 10).

Consequently, efforts in the coming year will focus on various risk-informed approaches to reduce the conservatism associated with the 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 more reactive 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 operating conditions (e.g., soluble boron concentration, moderator temperature, etc.) 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 are currently 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. An effective approach for validation of fission product worth is a key component for extension of burnup credit beyond actinide-only. Several techniques have been proposed for the quantification and validation of the worth of fission products in SNF. These techniques attempt to provide validation of cross sections for worth prediction, as well as validation of isotopic concentration predictions. Currently, two of these methods are being investigated. The first method uses the commercial reactor critical (CRC) database to include both concentration and cross section uncertainty components simultaneously. This method should validate most of the uncertainty components simultaneously, however, some isotopes of little importance during operation, e.g. ^{155}Gd would need additional validation measures. The second method produces a "guaranteed worth" by using factors to modify the predicted worth of individual fission products. These factors account for uncertainties in the worth predictions due to cross section and concentration uncertainties.

The current work to estimate a "guaranteed" 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. However, preliminary efforts in this area indicate that while the fission product worths vary from ~3 to 10 % in Δk, the guaranteed worths are about 1.5 to 5 %. The ratio of guaranteed worth to full worth as a function of burnup has been found to remain nearly constant with burnup. This suggests that the fractional contributions to the total fission product worth change only slightly with burnup. Hence, a single correction factor could possibly be applied to best-estimate worth values to obtain conservative worth estimates, regardless of the magnitude of the burnup.

All of the work discussed in this report 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.

Table 1. Nuclide sets used for analyses

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	O†

* Actinides are consistent with those specified in the DOE Topical Report (Ref. 7).
† 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.

Table 2. Number of axial burnup profiles (from the Yankee Atomic profile database[13]) that fall within various burnup and enrichment intervals

Upper bound of enrichment range (wt% ^{235}U)	Upper bound of burnup range (GWd/t)												Total
	5	10	15	20	25	30	35	40	45	50	55	60	
0.5	0	0	0	8	0	0	2	6	0	1	1	0	18†
1	0	0	0	0	0	0	0	0	0	0	0	0	0
1.5	0	0	3	2	3	0	0	0	0	0	0	0	8
2	1	10	29	31	16	1	0	0	0	0	0	0	88
2.5	0	3	37	49	16	12	1	0	0	0	0	0	118
3	8	88	79	165	109	127	110	28	5	0	1	0	720
3.5	0	45	103	171	124	114	209	163	69	20	0	0	1018
4	0	2	47	180	159	83	153	158	130	75	5	0	992
4.5	0	0	8	31	41	18	9	21	22	16	6	1	173
5	0	0	3	1	6	11	4	3	4	1	1	0	34
Total	9	148	309	638	474	366	488	379	230	113	14	1	3169

† These profiles had zero specified for their enrichment in the database (Ref. 13).

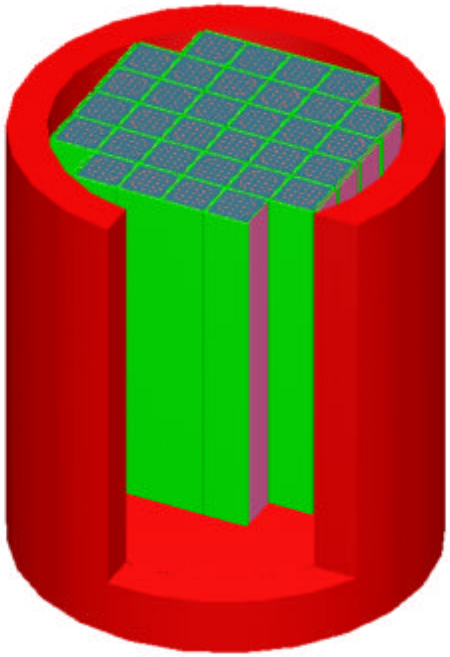

Figure 1. Cutaway view of GBC-32 burnup-credit cask model (one-half of full height).

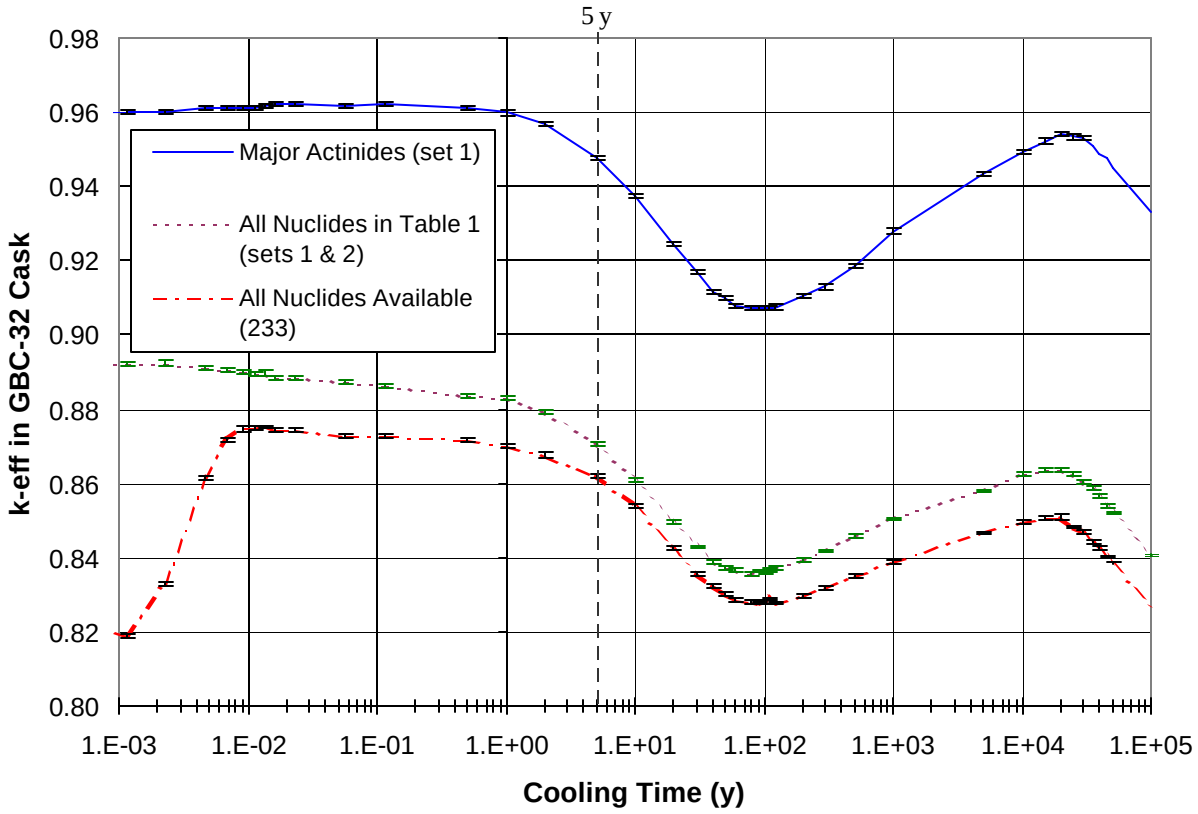

Figure 2. Values of k_{eff} in the GBC-32 cask as a function of cooling time for the three classifications of burnup credit (burnup-credit classifications are defined in Table 1). The results correspond to fuel with 4.0 wt% ^{235}U initial enrichment that has accumulated a 40 GWd/t burnup.

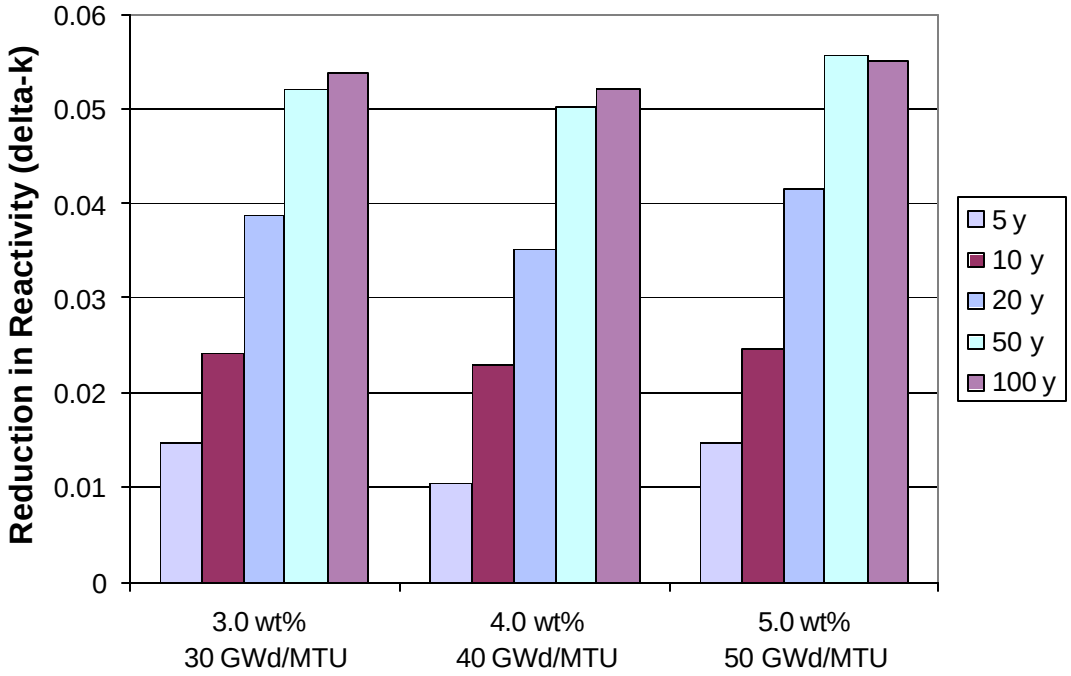

Figure 3. Reactivity reduction as a function of cooling time for some typical initial enrichment and burnup combinations with actinide-only burnup credit.

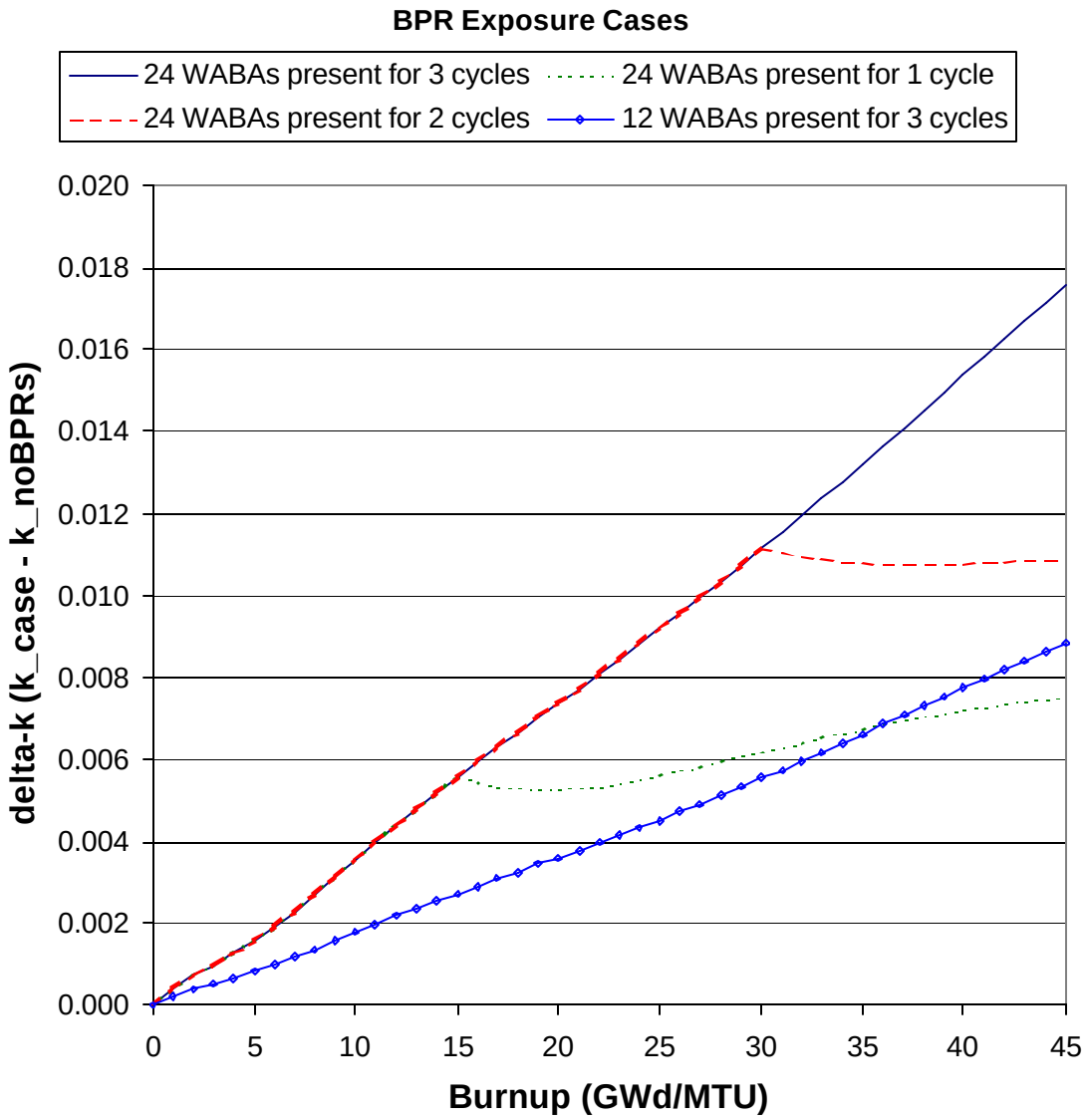

Figure 4. D**k values (relative to the no-BPR condition) as a function of burnup for various BPR exposures using the actinide-only assumption. Results correspond to 4.0 wt% ^{235}U initial enrichment that has been exposed to Westinghouse WABA rods (three cycles of 15 GWd/t burnup per cycle were assumed).**

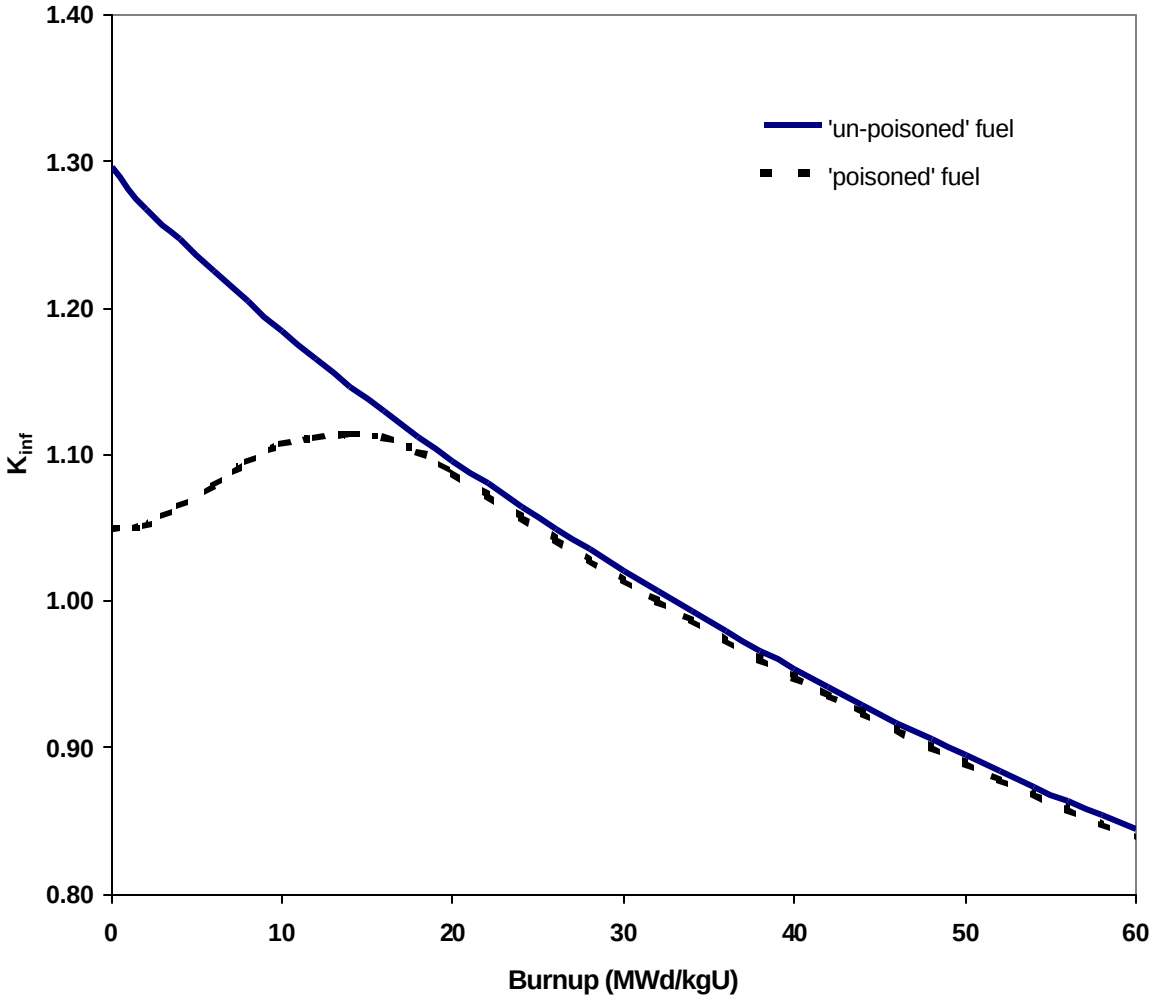

Figure 5. Typical reactivity behavior of PWR fuel with and without IBAs (neutron poisons) present.

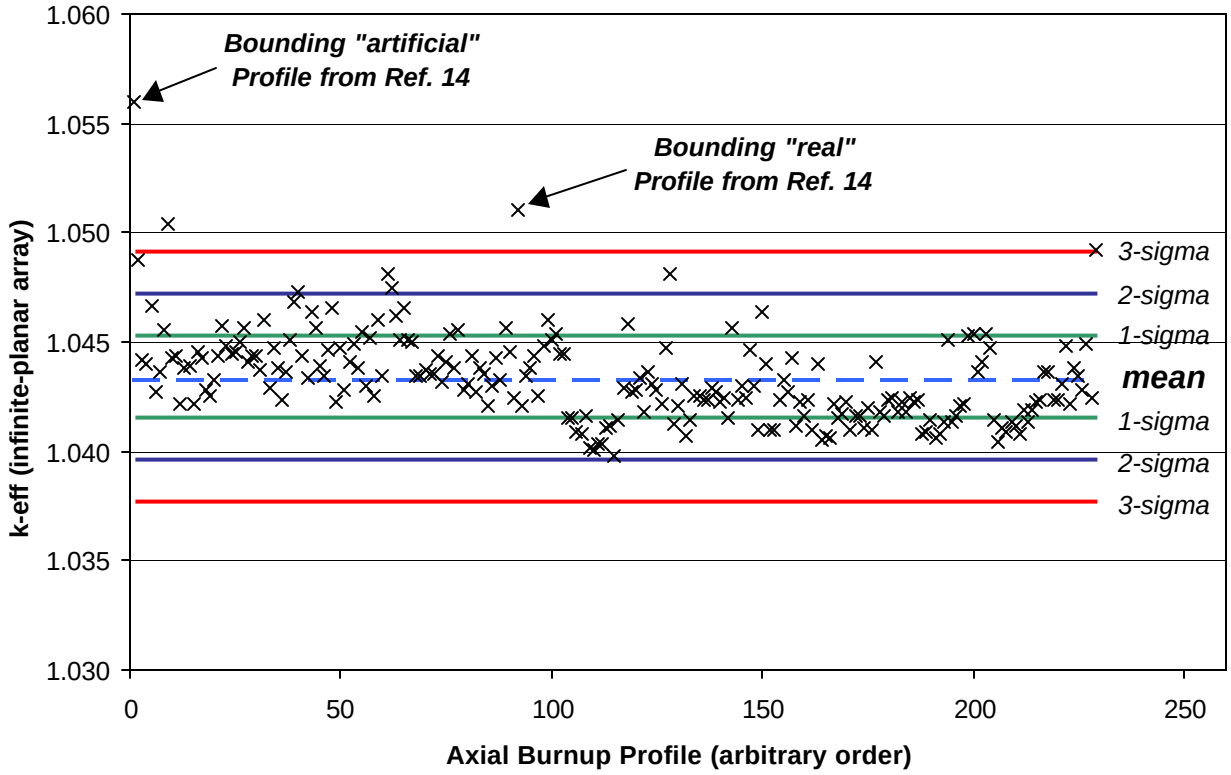

Figure 6. k_{eff} values based on database axial burnup profiles for the burnup range 38-42 GWd/t.

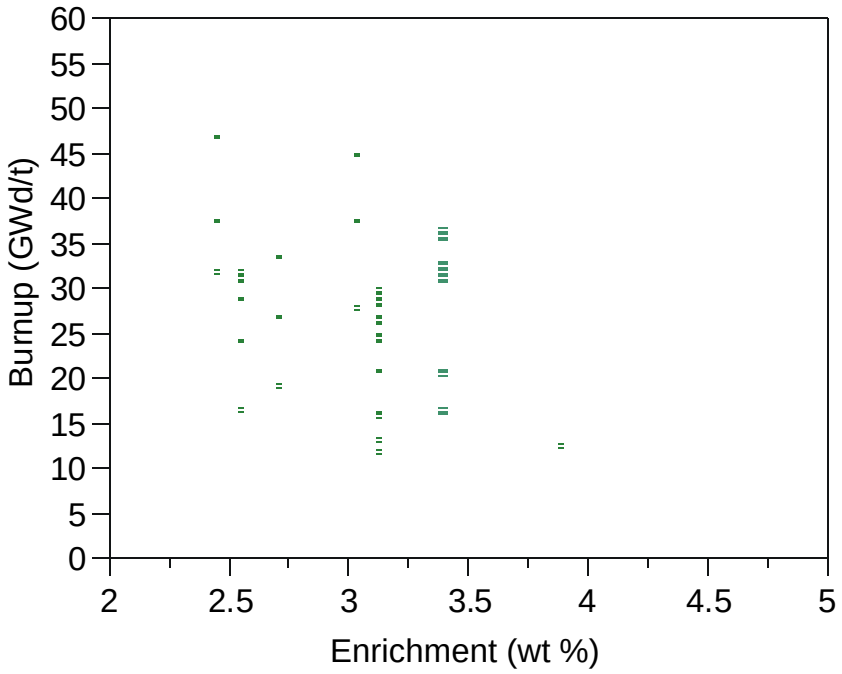

Figure 7. Enrichment and burnup of 46 PWR assay samples used in recent validation studies.

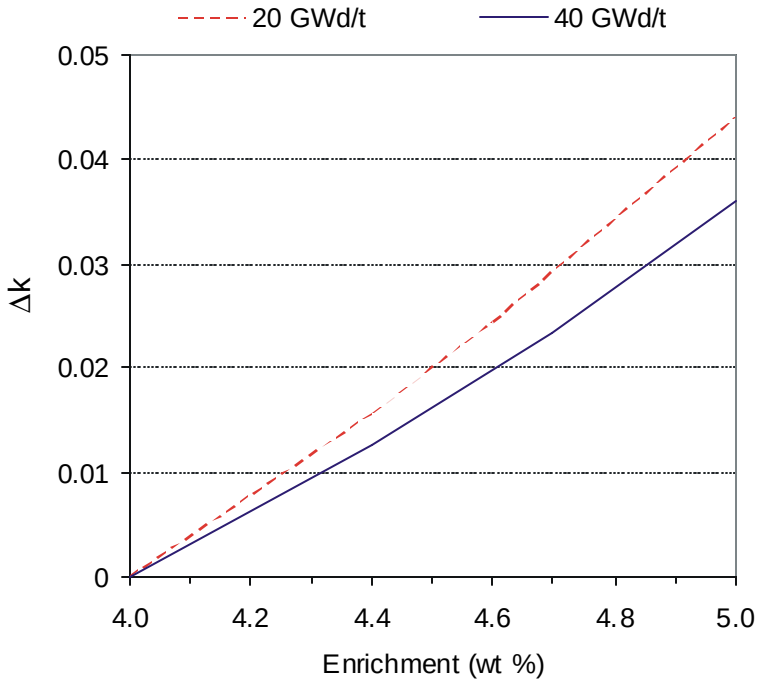

Figure 8. ISG-8r1 loading offset Dk for the GBC-32 cask design.

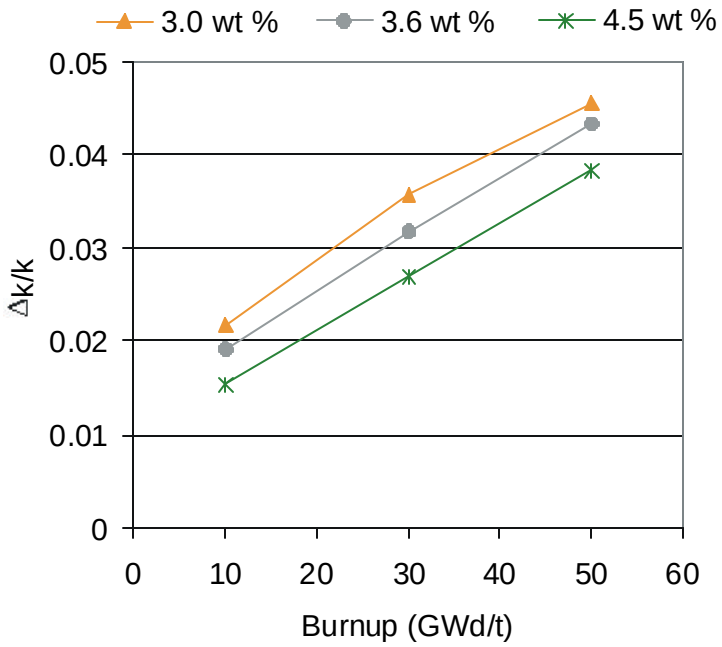

Figure 9. Reactivity penalty associated with actinide uncertainties as a function of burnup.

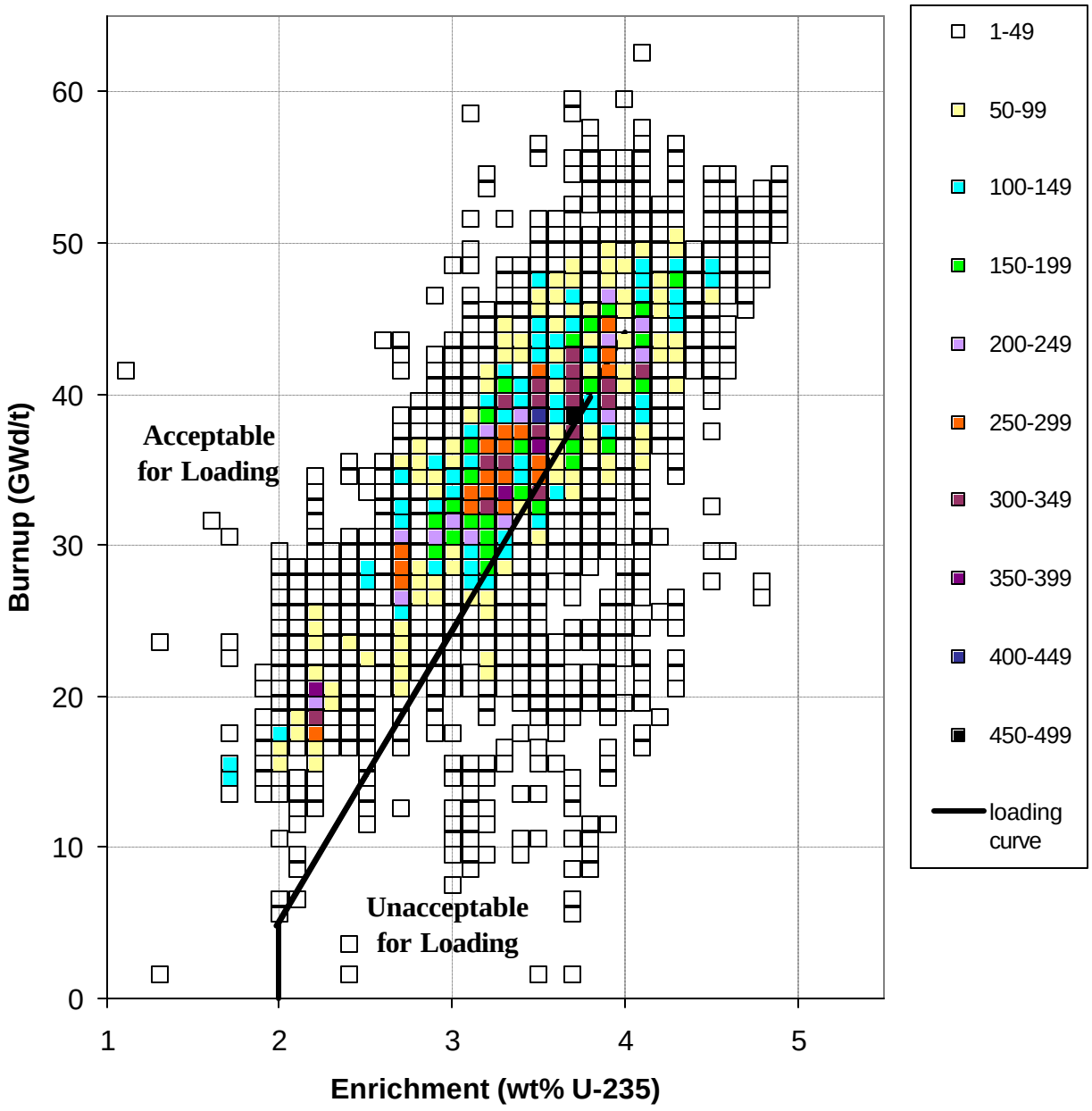

Figure 10. PWR SNF discharge data through 1998 (numbers in legend indicate number of assemblies), shown with illustrative loading curve for GBC-32 cask.

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