ATTACHMENT 6A TECHNICAL RECOMMENDATIONS FOR THE CRITICALITY SAFETY REVIEW OF PRESSURIZED-WATER REACTOR SPENT NUCLEAR FUEL TRANSPORTATION PACKAGES AND STORAGE CASKS THAT USE BURNUP CREDIT

6A.1 Introduction

The overall reactivity decrease of nuclear fuel irradiated in light-water reactors occurs because of the combined effect of the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides (non-fissile actinides and fission products). Burnup credit refers to accounting for partial or full reduction of spent nuclear fuel (SNF) reactivity caused by irradiation. Section 6.4.7 of this standard review plan (SRP) provides guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for use in reviewing commercial light-water reactor SNF package designs that seek burnup credit. This attachment provides the technical bases for the burnup-credit recommendations provided in the SRP and for SNF dry storage; thus, the attachment discusses both storage and transportation.

Historically, criticality safety analyses for transportation and dry cask storage of SNF assumed the fuel contents to be unirradiated ("fresh") fuel. In September 2002, the NRC Spent Fuel Project Office (SFPO) issued Interim Staff Guidance (ISG)-8, Revision 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," to provide recommendations for the use of actinide-only burnup credit (i.e., burnup credit using only major actinide nuclides) in storage and transport of pressurized-water reactor (PWR) SNF. Based on the data available for burnup-credit depletion and criticality computer code validation at the time ISG-8, Revision 2, was issued, SFPO staff recommended actinide-only credit. Additionally, the staff recommended that a measurement be performed to confirm the reactor record burnup value for SNF assemblies to be stored or transported in storage cask or package designs that credit burnup in the criticality analysis.

Since ISG-8, Revision 2, was issued, significant progress has been made in research on the technical and implementation aspects of burnup credit, with the support of the NRC Division of Spent Fuel Storage and Transportation (formerly SFPO), by the NRC Office of Nuclear Regulatory Research (RES), and its contractors at Oak Ridge National Laboratory (ORNL). This attachment summarizes the findings of a number of reports and papers published as part of the research program RES directed over the last several years. It is recommended that the staff read the referenced reports and papers to understand the detailed evaluation of specific burnup-credit parameters discussed in this attachment. A comprehensive bibliography of burnup-credit-related technical reports and papers is provided at http://www.ornl.gov/sci/nsed/rnsd/pubs_burnup.shtml.

6A.2 General Approach in Safety Analysis

Criticality safety analyses of SNF storage or transportation systems are complex in terms of both the computer modeling of the system and the required fuel information. The assumption of unirradiated fuel at maximum initial enrichment provides a straightforward approach for the criticality safety analysis of an SNF dry storage or transportation system. This is a conservative approach to criticality safety and limits the system capacity. In comparison to the fresh fuel assumption, criticality safety analyses for SNF systems that credit burnup require the following:

- additional information and assumptions for input to the analysis
- additional analyses to obtain the SNF compositions
- additional validation efforts for the depletion and decay software
- enhanced validation to address the additional nuclides in the criticality analyses
- verification that the fuel assembly to be loaded meets the minimum burnup requirements made before loading the system

The use of burnup credit for SNF storage casks and transportation packages provides for increased fuel capacities and higher limits on allowable initial enrichments for such systems. Applications for PWR SNF storage cask and transportation package certificates of compliance (CoCs) have generally shifted to high-capacity designs (i.e., 32 fuel assemblies or greater) in the past 15 years. To fit this many assemblies in a similarly sized SNF system, applicants have removed flux traps present in lower capacity designs (i.e., 24 fuel assemblies or less) and replaced them with single-neutron-absorber plates between assemblies. Flux traps consist of two neutron-absorber plates separated by a water region, with the water serving to slow neutrons for more effective absorption. Single-neutron-absorber plates are less effective absorbers than flux trap designs and result in a system that cannot be shown to be subcritical in unborated water without the use of some level of burnup credit.

An important outcome from a burnup credit criticality safety analysis is an SNF loading curve, showing the minimum burnup required for loading as a function of initial enrichment and cooling time. For a given system loading of SNF, the effective neutron multiplication factor (k_{eff}) will increase with higher initial enrichments, decrease with increases in burnup, and decrease with cooling time from 1 year to approximately 100 years. Information that should be considered in specifying the technical limits for fuel acceptable for loading includes fuel design, initial enrichment, burnup, cooling time, and the reactor conditions under which the fuel is irradiated. Thus, depending on the assumptions and approach used in the safety analysis and the limiting k_{eff} criterion, a loading curve or set of loading curves can be generated to define the boundaries between acceptable and unacceptable SNF specifications for system loading.

The recommendations in Section 6.4.7 of this SRP chapter include the following:

- general information on limits for the certification basis
- recommended assumptions regarding reactor operating conditions
- guidance on code validation with respect to the isotopic depletion evaluation
- guidance on code validation with respect to the k_{eff} evaluation
- guidance on preparation of loading curves and the process for assigning a burnup loading value to an assembly

A criticality safety analysis that uses burnup credit should consider each of these five areas.

The five recommendations listed above were developed with intact fuel as the basis. Extending the recommendations to fuel that is not intact may be warranted if the applicant can demonstrate

that any additional uncertainties associated with the irradiation history and structural integrity (both during and subsequent to irradiation) of the fuel assembly have been addressed. In particular, a model that bounds the uncertainties associated with the allowed fuel inventory and fuel configuration in the system should be applied. Such a model should include the selection of appropriate burnup distributions and any potential rearrangement of fuel that is not intact during normal and accident conditions. The applicant should also apply each of the recommendations in this review guidance and justify any exceptions taken because of the nature of the fuel (e.g., the use of an axial profile that is not consistent with the recommendation). Section 7.4.14 of this SRP provides guidance for classifying the condition of the fuel (e.g., damaged, intact) for SNF transportation.

The validation methods presented in Sections 4 and 5 of this attachment were performed for a representative storage cask/transportation package model, known as the generic burnup-credit cask (GBC)-32, described in NUREG/CR-6747, "Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit," issued October 2001. As this attachment will discuss later, to directly use bias and bias uncertainty numbers developed in NUREG/CR-7108, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions," issued April 2012, and NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions," issued April 2012, applicants must use the same isotopic depletion and criticality code and nuclear data as were used in the isotopic depletion and criticality validation performed in those reports. Additionally, applicants must demonstrate that their SNF storage or transportation system design is similar to the GBC-32 used to develop the validation methodologies in NUREG/CR-7108 and NUREG/CR-7109. This demonstration should consist of a comparison of system materials and geometry, including neutron-absorber material and dimensions, assembly spacing, and reflector materials and dimensions. This demonstration should also include a comparison of neutronic characteristics such as hydrogen-to-fissile atom ratios (H/X), energy of average neutron lethargy-causing fission (EALF), neutron spectra, and neutron reaction rates. Applicability of the validation methodology to systems with characteristics that deviate substantially from those for the GBC-32 should be justified. Sensitivity and uncertainty analysis tools, such as those provided in the SCALE code system, can provide a quantitative comparison of the GBC-32 to the application of interest.

The recommendations in this review guidance were developed with PWR fuel as the basis. Typically, dry storage and transportation applicants have not sought boiling-water reactor (BWR) burnup credit, because of the complexity of the fuel and irradiation parameters, the lack of code validation data to support burnup credit, and a general lack of need for such credit in existing designs. The NRC has initiated a research project to obtain the technical basis for BWR burnup credit. BWR fuel assemblies typically have neutron-absorbing material, typically gadolinium oxide (Gd_2O_3), mixed in with the uranium oxide of the fuel pellets in some rods. This neutron absorber depletes more rapidly than the fuel during the initial parts of its irradiation, which causes the fuel assembly reactivity to increase and reach a maximum value at an assembly-average burnup typically less than 20 gigawatt-days per metric ton of uranium (GWd/MTU). Then reactivity decreases for the remainder of fuel assembly irradiation. Criticality analyses of BWR SNF pools typically employ what are known as "peak reactivity" methods to account for this behavior. NUREG/CR-7194, "Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems," issued April 2015, reviews several existing peak-reactivity methods and demonstrates that a conservative set of analysis conditions can be identified and implemented to allow criticality safety analysis of BWR SNF assemblies at

peak reactivity in storage and transportation systems. The reviewer should consult NUREG/CR-7194 if the applicant uses peak reactivity BWR burnup-credit methods in its criticality analysis.

This SRP does not address credit for BWR burnup beyond peak reactivity. An NRC research program is currently investigating methods for conservatively including such credit in a BWR criticality analysis for SNF storage systems and transportation packages, but at this time, the NRC does not recommend burnup credit beyond peak reactivity. The reviewer should consider conservative analyses of BWR burnup credit beyond peak reactivity on a case-by-case basis, consulting the latest research results in this area (i.e., NRC Letter Reports, NUREG/CRs).

The remainder of this attachment discusses recommendations in each of the five burnup-credit areas and provides technical information and references that should be considered in the review of the application.

6A.3 Limits for Certification/Licensing Basis (Section 6.4.7.1 of this SRP)

Available validation data support actinide-only and actinide and fission product burnup credit for uranium dioxide (UO_2) fuel, enriched up to 5.0 weight percent uranium-235, that is irradiated in a PWR to an assembly-average burnup value up to 60 GWd/MTU and cooled out of the reactor between 1 and 40 years.

Nuclides of Importance

Several studies have been performed to identify the nuclides that have the most significant effect on the calculated value of k_{eff} as a function of burnup and cooling time. These results are summarized in NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," issued February 2000. This report concludes that the actinides and fission products listed in Tables 6A-1 and 6A-2 are candidates for inclusion in burnup-credit analyses for storage and transportation systems, based on their relative reactivity worth at the cooling times of interest.

The relative reactivity worth of the nuclides will vary somewhat with fuel design, initial enrichment, and cooling time, but the important nuclides (fissile nuclides and select nonfissile absorbers) remain the same and have been substantiated by many independent studies. These nuclides have the largest impact on k_{eff}, and there is a sufficient quantity of applicable experimental data available for validation of the analysis methods, as Sections 5 and 6 of this attachment discuss. Accurate prediction of the concentrations for the nuclides in Tables 6A-1 and 6A-2 requires that the depletion and decay calculations include nuclides beyond those listed in the tables. Additional actinides and fission products are needed to ensure that the transmutation chains and decay chains are accurately handled. Methods are also needed to accurately simulate the influence of the fission product compositions on the neutron spectrum, which in turn impacts the burnup-dependent cross sections. To accurately predict the reactivity effect of fission products, explicit representation of the important fission product transmutation and decay chains is needed to obtain the individual fission product compositions.

Applicants attempting to credit neutron-absorbing isotopes other than those listed in these tables should ensure that such isotopes are nonvolatile, nongaseous, and relatively stable and should provide analyses to determine the additional depletion and criticality code bias and bias uncertainty associated with these isotopes. These analyses should be accompanied by additional relevant critical experiment and radiochemical assay (RCA) data, to the extent practicable, or include sufficient penalties to account for the lack of such data.

Table 6A-1 Recommended set of nuclides for actinide-only burnup credit		
^{234}U	^{235}U	^{238}U
^{238}Pu	^{239}Pu	^{240}Pu
^{241}Pu	^{242}Pu	^{241}Am

Table 6A-2 Recommended set of additional nuclides for actinide and fission product burnup credit			
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{147}Sm	^{149}Sm
^{150}Sm	^{151}Sm	^{152}Sm	^{143}Nd
^{145}Nd	^{151}Eu	^{153}Eu	^{155}Gd
^{236}U	^{243}Am	^{237}Np	

Burnup and Enrichment Limits

NUREG/CR-7108 demonstrates that the range of existing RCA data that are readily available for validation extends up to 60 GWd/MTU and 4.657 weight percent uranium-235 initial enrichment. Though limited RCA data are available above 50 GWd/MTU, it is the staff's judgement that credit can reasonably be extended up to 60 GWd/MTU. Credit should not be extended to assembly-average burnups beyond this level, though local burnups can be higher. Fuel with an assembly-average burnup greater than 60 GWd/MTU can be loaded into a burnup-credit system, but credit should be taken only for the reactivity reduction up to 60 GWd/MTU. Additionally, while the enrichment range covered by the available assay data has increased, it has not increased enough to warrant a change in the maximum initial enrichment that can be considered in a burnup credit analysis; thus, the initial enrichment limit for the licensing or certification basis remains at 5.0 weight percent uranium-235.

Cooling Time

Figure 6A-1 illustrates the expected reactivity behavior for SNF in a hypothetical GBC-32 system for an analysis using major actinide concentrations and various actinide and fission product concentrations in the calculation of k_{eff}. Reactivity begins to rise around 100 years after discharge, which means the timeframe for interim SNF storage should be considered in the evaluation of acceptable cooling times. The curve indicates that the reactivity of the fuel at 40 years is about the same as that of fuel cooled to 200 years. The Commission has instructed staff to review the regulatory programs for SNF storage and transportation, considering extended storage beyond 120 years (NRC 2010). In light of the increasingly likely scenario of extended dry storage of SNF, the CoC for an SNF transportation package or the CoC or license for dry storage may require an additional condition for the applicability of the credited burnup of the SNF contents. The condition would depend on the type of credit taken and the post-irradiation decay time credited in the analysis. For example, crediting 40 years would result in a CoC condition limiting the applicability of the credited burnup to 160 years after fuel discharge. Approval of a cooling time longer than 5 years for burnup credit in dry storage or transportation systems does not automatically guarantee acceptance for disposal without repackaging. NUREG/CR-6781, "Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses," issued January 2003, provides a comprehensive study of the effect of cooling time on burnup credit for various package designs and SNF compositions.

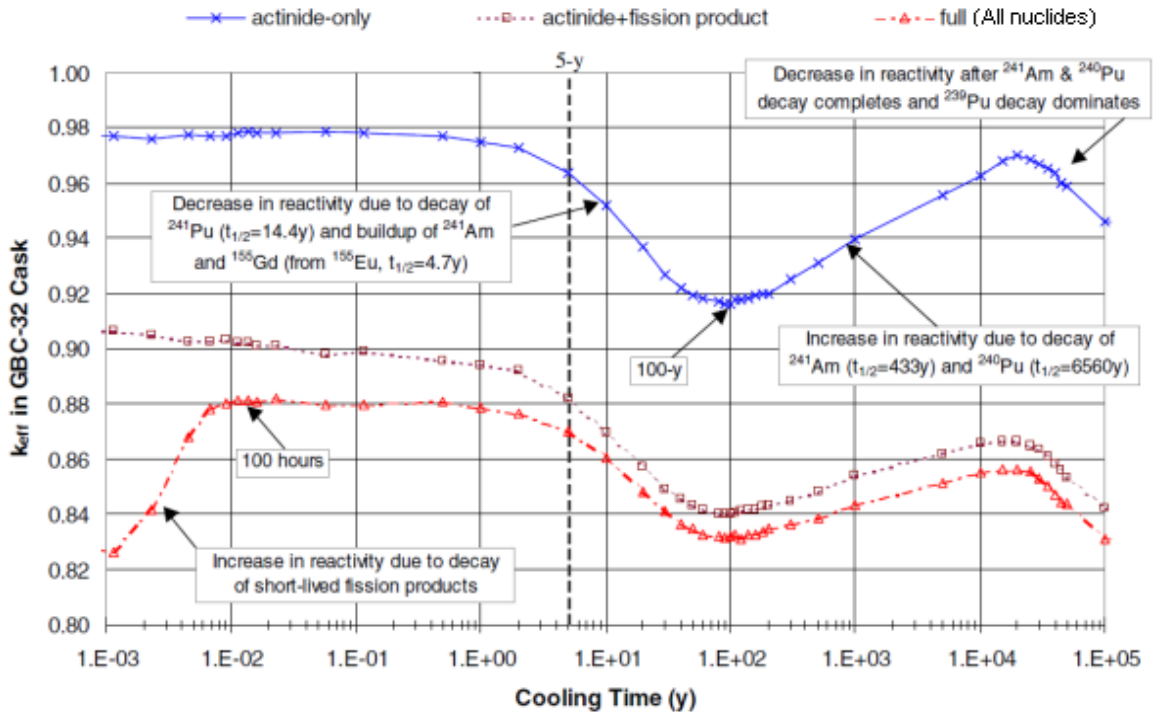

Figure 6A-1 Reactivity behavior in the GBC-32 cask as a function of cooling time for fuel with 4.0 weight percent uranium-235 initial enrichment and 40 Gwd/MTU burnup (Source: NRC 2010)

Summary

The acceptance criteria for burnup credit are based on the characteristics of SNF discharged to date, the parameter ranges considered in most technical investigations, and the experimental data available to support development of a calculational bias and bias uncertainty. As indicated, a safety analysis that uses parameter values outside those recommended by the SRP should (i) demonstrate that the measurement or experimental data necessary for proper code validation have been included and (ii) provide adequate justification that the analysis assumptions or the associated bias and bias uncertainty have been established in a way that bounds the potential impacts of limited measurement or experimental data. Even within the recommended range of parameter values, the reviewer should exercise care in assessing whether the analytic methods and assumptions used are appropriate, especially near the ends of the range.

6A.4 Model Assumptions (Section 6.4.7.2 of this SRP)

The actinide and fission product compositions used to determine a value of k_{eff} should be calculated using fuel design and reactor operating parameter values that encompass the range of design and operating conditions for the proposed contents. The proposed contents may consist of the entire population of discharged PWR fuel assemblies, a specific design of PWR fuel assembly [e.g., W17x17 optimized fuel assembly (OFA)], or a smaller, specific population from a particular site. The k_{eff} value should be calculated using package models, analysis assumptions, and code inputs that allow accurate representation of the physics in the system. The following discusses important parameters that should be addressed in depletion analyses and k_{eff} calculations in a burnup-credit evaluation.

Reactor Operating History and Parameter Values

Section 4.2 of NUREG/CR-6665 discusses the impacts of fuel temperature, moderator temperature and density, soluble boron concentration, specific power and operating history, and burnable absorbers on the k_{eff} of SNF in a storage cask or transportation package.

As the assumed fuel temperature used in the depletion model increases, the k_{eff} for the SNF in the storage cask or package will increase. The k_{eff} will also increase with increases in either moderator temperature (lower density) or the soluble boron concentration. Analyses for both actinide-only and actinide-plus-fission product evaluations exhibit these trends in k_{eff}. Figures 6A-2 to 6A-4 provide examples of the Δk impact seen from differences in fuel temperature, moderator temperature, and soluble boron concentration. The system modeled to determine these results was an infinite array of storage cells, but similar results have been confirmed for finite, reflected systems. All of these increases are the result of the parameter increase causing increased production of fissile plutonium nuclides and decreased uranium-235 utilization.

Specific power and operating history have a much more complex impact but a very small effect on the storage cask or package k_{eff} value. Figures 6A-5 and 6A-6 show the variation of k_{inf} with specific power for various initial enrichment and burnup combinations, for actinide-only and actinide-plus-fission product burnup credit, respectively. Irradiation at higher specific power results in a slightly higher k_{eff} for actinide-only burnup credit, but the reverse is true for burnup credit that includes actinides and fission products (see Section 3.4.2.3 of DeHart 1996). Although the specific power at the end of irradiation is most important, the assumption of constant full power is more straightforward and acceptable, while having minimal impact on the k_{eff} value relative to other assumptions.

NUREG/CR-6665 and DeHart (1996) provide more detailed information on the impact of each parameter or phenomenon that should be assumed in the depletion model. Independent studies have substantiated each of the trends and impacts. However, to model the irradiation of the fuel to produce bounding values for k_{eff} consistent with realistic reactor operating conditions, information is needed on the range of actual reactor conditions for the proposed SNF to be loaded in a package. Loading limitations tied to the actual operating conditions will be needed unless the operating condition values assumed in the model can be justified as those that produce the maximum k_{eff} values for the anticipated SNF package contents. As illustrated by the case of specific power and operating history, the bounding conditions and appropriate limitations may differ for actinide-only burnup credit versus actinide-plus-fission-product burnup credit, since the parameter impact may trend differently for these two types of burnup credit. The sensitivity to variations in the depletion parameter assumptions differs for the two types of burnup credit, with actinide-plus-fission-product burnup credit analyses exhibiting greater sensitivity for some parameters (see NUREG/CR-6800, “Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs,” issued March 2003).

Also, the most reactive fuel design before irradiation will not necessarily have the highest reactivity after discharge from the reactor, and the most reactive fuel design may differ at various burnup levels. Thus, if various fuel designs are to be allowed in a particular package design, parametric studies should be performed to demonstrate the most reactive SNF design for the range of burnup and enrichments considered in the safety analysis. Another option is to provide loading curves for each fuel assembly design and allow only one assembly type in each package loading.

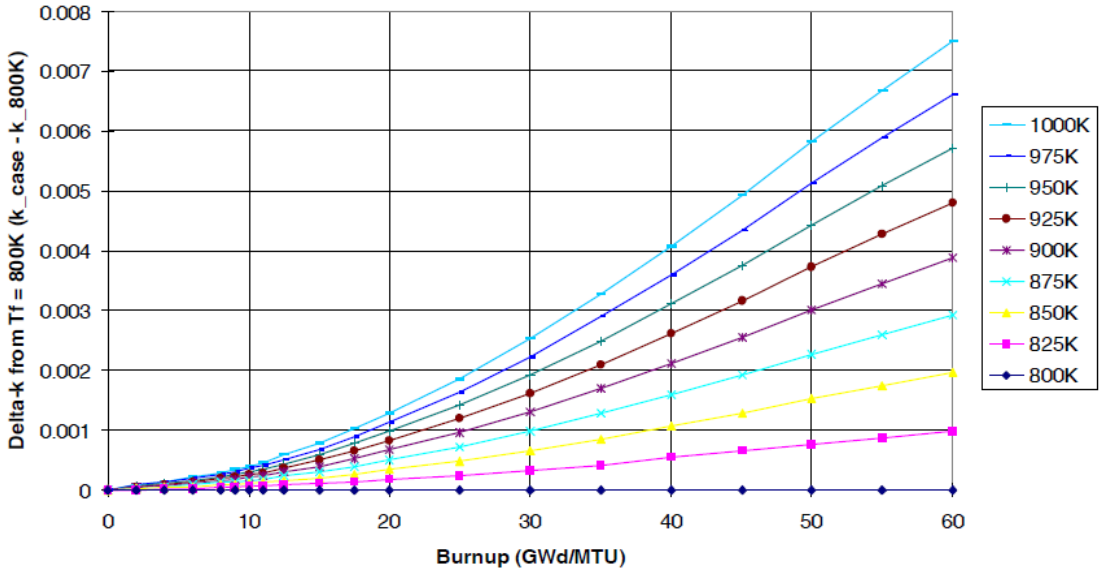

Figure 6A-2 Reactivity effect of fuel temperature during depletion on K_{inf} in an array of poisoned storage cells; results correspond to fuel with 5.0 weight percent initial uranium-235 enrichment (Source: Withee 2002)

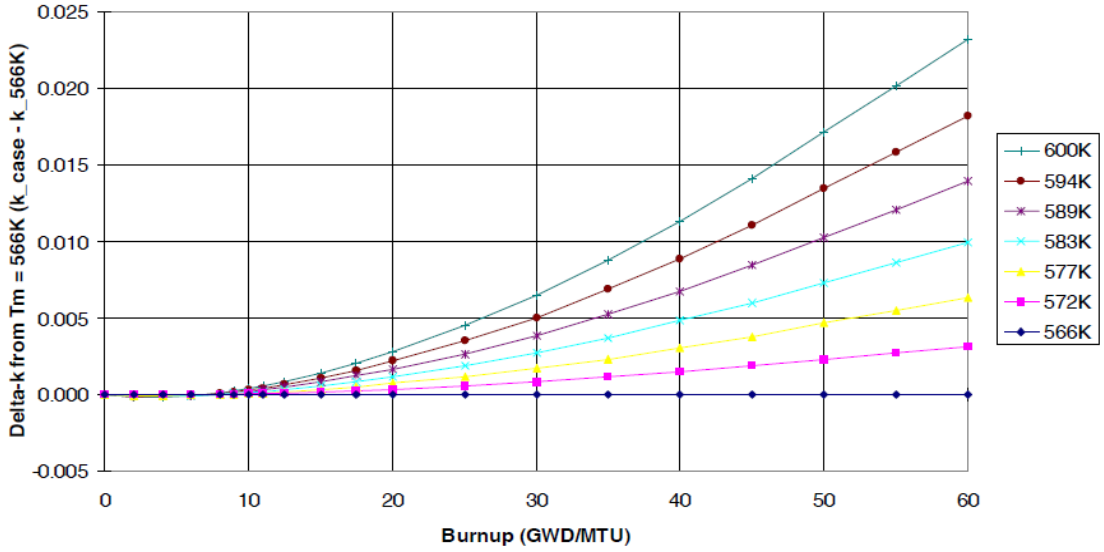

Figure 6A-3 Reactivity effect of moderator temperature during depletion on K_{inf} in an array of poisoned storage cells; results correspond to fuel with 5.0 weight percent initial uranium-235 enrichment (Source: Withee 2002)

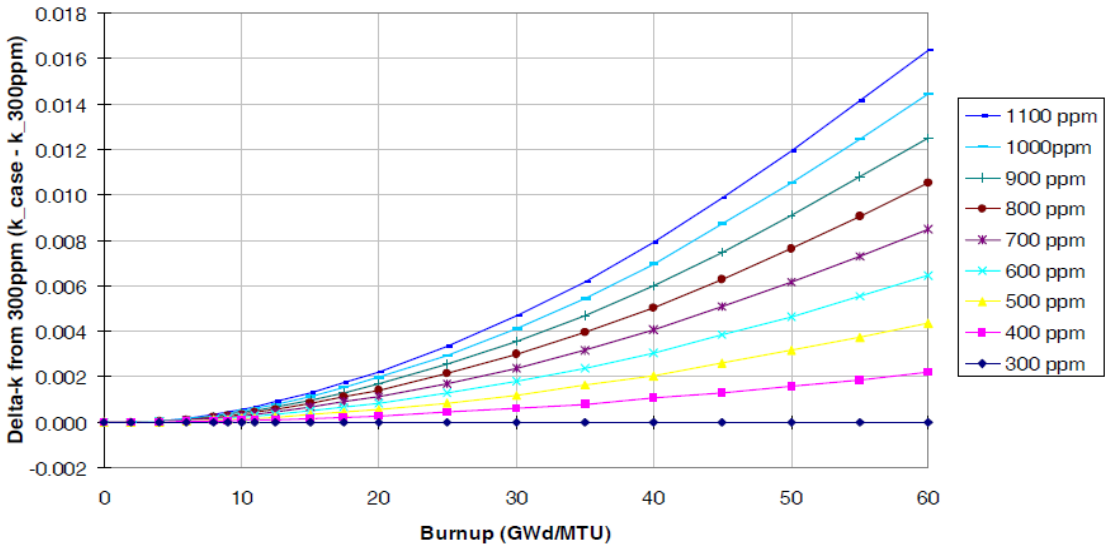

Figure 6A-4 Reactivity effect of soluble boron concentration during depletion on K_{inf} in an array of poisoned storage cells; results correspond to fuel with 5.0 weight percent initial uranium-235 enrichment (Source: Withee 2002)

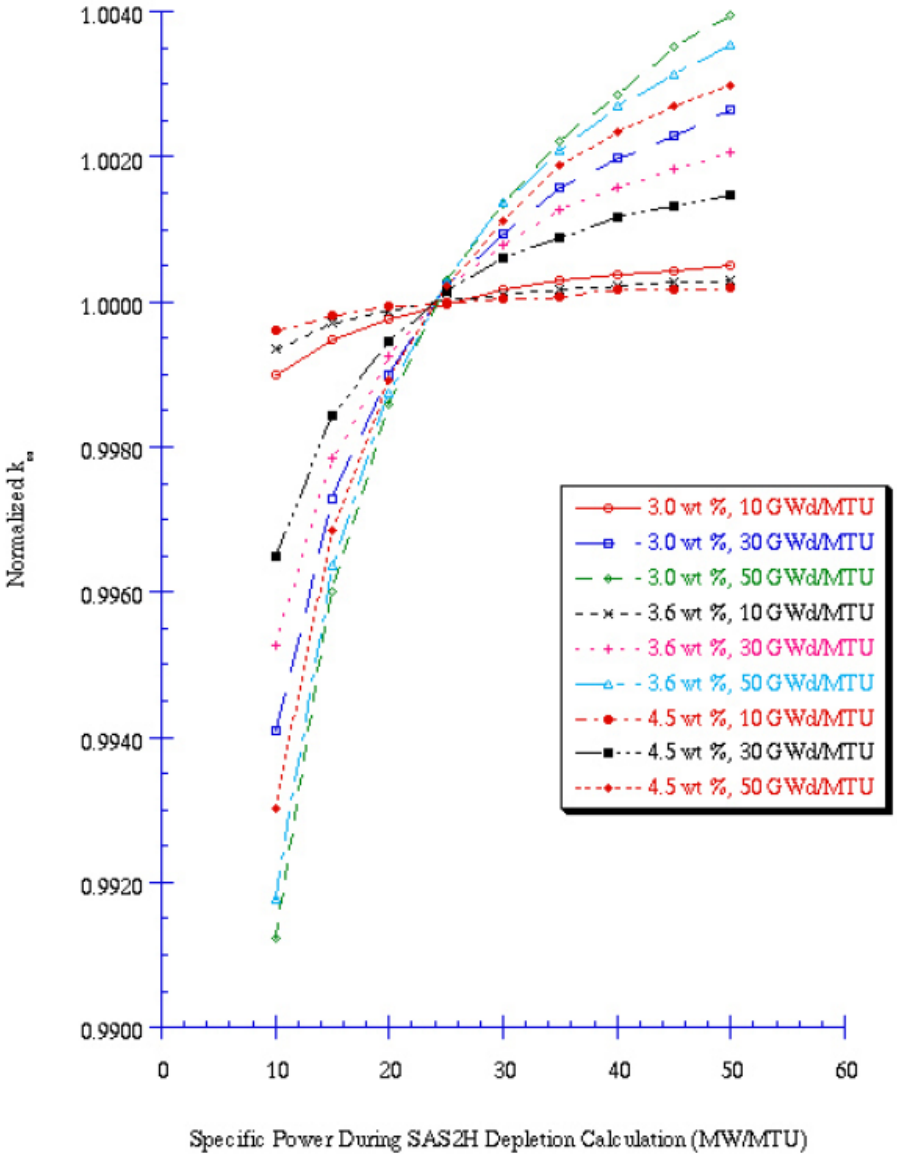

Figure 6A-5 Reactivity effect of specific power during depletion on K_{inf} in an array of fuel pins (actinides only) (Source: Dehart 1996)

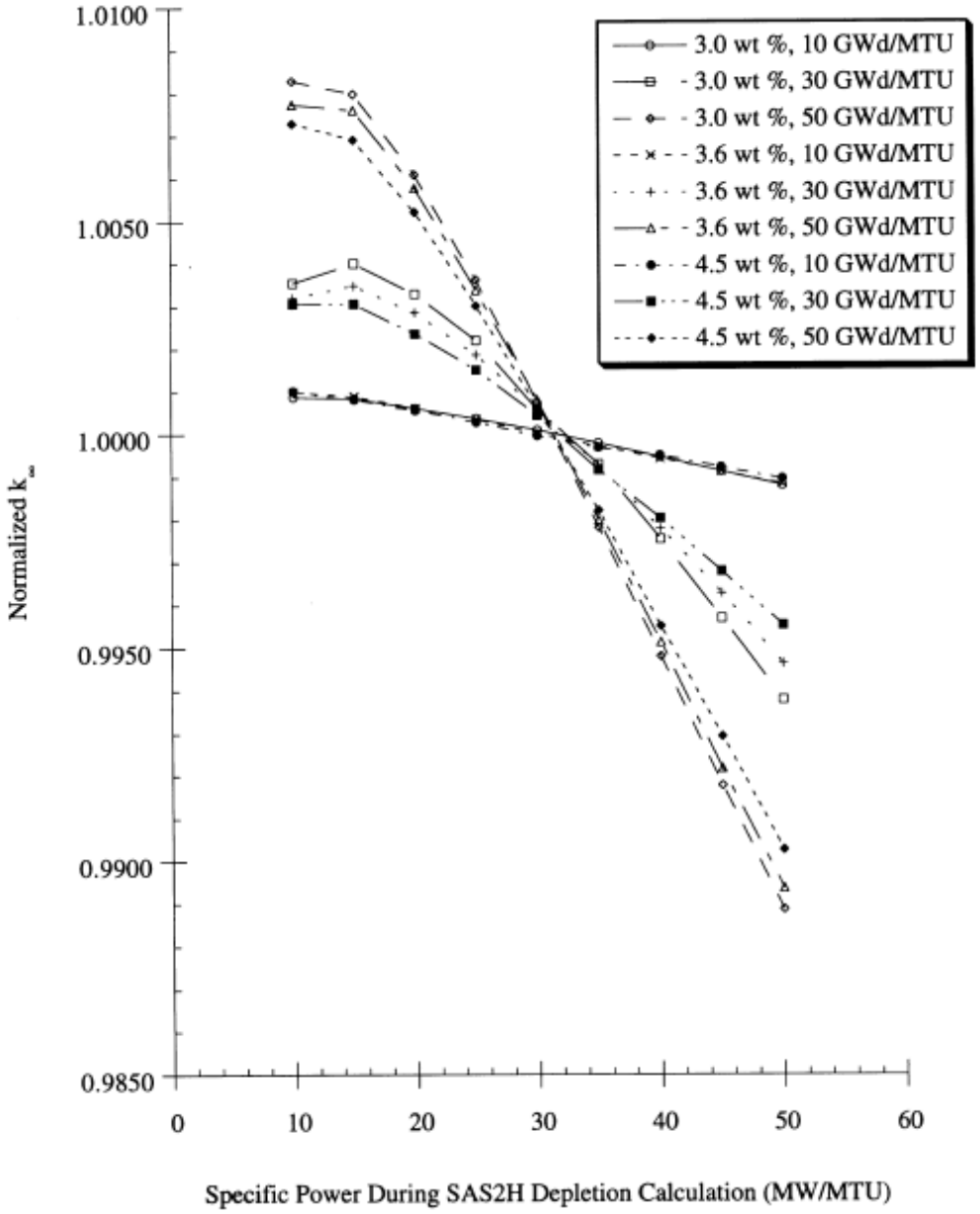

Figure 6A-6 Reactivity effect of specific power during depletion on K_{inf} in an array of fuel pins (actinides and fission products) (Source: Dehart 1996)

Horizontal Burnup Profiles

Consideration of pin-by-pin burnups (and associated variations in SNF composition) does not appear to be necessary for analysis of the integral k_{eff} value in a SNF storage cask or package. To date, PWR cores have been managed such that the vast majority of assemblies experience a generally uniform burnup horizontally across the assembly during an operating cycle. However, assemblies on the periphery of the core may have a significant variation in horizontal burnup after a cycle of operation (see DOE/RW-0496, "Horizontal Burnup Gradient Datafile for PWR Assemblies," issued May 1997). In large storage casks or rail packages, the probability that underburned quadrants of multiple fuel assemblies will be oriented in such a way as to have a substantial impact on k_{eff} is not expected to be significant. However, for smaller systems, the

effect can be significant. The safety evaluation should address the impact of horizontal burnup gradients (such as found in DOE/RW-0496) on their package design or demonstrate that the assemblies to be loaded in the package will be verified to not have such gradients. One acceptable approach would be to determine the difference in k_{eff} for a package loaded with fuel having a horizontal burnup gradient and a package loaded with the same fuel having a uniform horizontal burnup (i.e., no gradient). The fuel with the gradient would be arranged so as to maximize the reactivity effect of the gradient. The reactivity difference between the two cases could then be applied to the remaining analyses.

Axial Burnup Profiles

Considerable attention should be paid to the axial burnup profile(s) selected for use in the safety evaluation. A uniform axial profile is generally bounding at low burnups but is increasingly nonconservative at higher burnups because of the increasing relative worth of the fuel ends, as demonstrated in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," issued March 2003. Figure 6A-7 illustrates an example of this phenomenon for an actinide-only burnup credit analysis. As the figure shows, a uniform axial profile was conservative for that analysis at burnups less than about 20 GWd/MTU, but nonconservative at higher burnups. The burnup range at which this transition occurs will vary with fuel design and the type of burnup credit.

Section 6.4.7.2 of this SRP and this attachment indicate that any analysis should provide an "accurate representation of the physics" in the system (i.e., the package, including package arrays). Thus, the applicant should select and model the axial burnup profile(s) in the analyses (including an appropriate number of axial material zones) that encompass the proposed contents and their range of potential k_{eff} values. The applicant should account for variance of the axial effect with burnup, cooling time, SNF nuclides used in the prediction of k_{eff}, and package design. The reviewer should consider the range of profiles anticipated for the fuel to be loaded in the package.

The publicly available database of axial profiles in YAEC-1937, "Axial Burnup Profile Database for Pressurized Water Reactors," issued May 1997, is recommended as an appropriate source for selecting axial burnup profiles that will encompass the SNF anticipated for loading in a burnup credit package. While the database represents only 4 percent of the assemblies discharged through 1994, NUREG/CR-6801 indicates that it provides a representative sampling of discharged assemblies. This conclusion is reached on the basis of fuel vendor/reactor design, types of operation (i.e., first cycles, out-in fuel management, and low-leakage fuel management), burnup and enrichment ranges, use of burnable absorbers (including different absorber types), and exposure to control rods (CRs) [including axial power shaping rods (APSRs)]. NUREG/CR-6801 also indicates that while the database has limited data for burnup values greater than 40 GWd/MTU and initial enrichments greater than 4.0 weight percent uranium-235, there is a high probability that the profiles resulting in the highest reactivity at intermediate burnup values will yield the highest reactivity at higher burnups. Thus, the existing database should be adequate for burnups beyond 40 GWd/MTU and initial enrichments above 4.0 weight percent uranium-235 if profiles are selected that include a margin for the potential added uncertainty in moving to the higher burnups and initial enrichments allowed in Section 6.4.7.1 of this SRP chapter and Section 3 of this attachment. Given the limited nature of the database, NUREG/CR-6801 includes an evaluation of the database's limiting profiles and the impacts of loading significantly more reactive assemblies in the place of assemblies with limiting profiles. NUREG/CR-6801 concludes that, based on the low consequence of the more

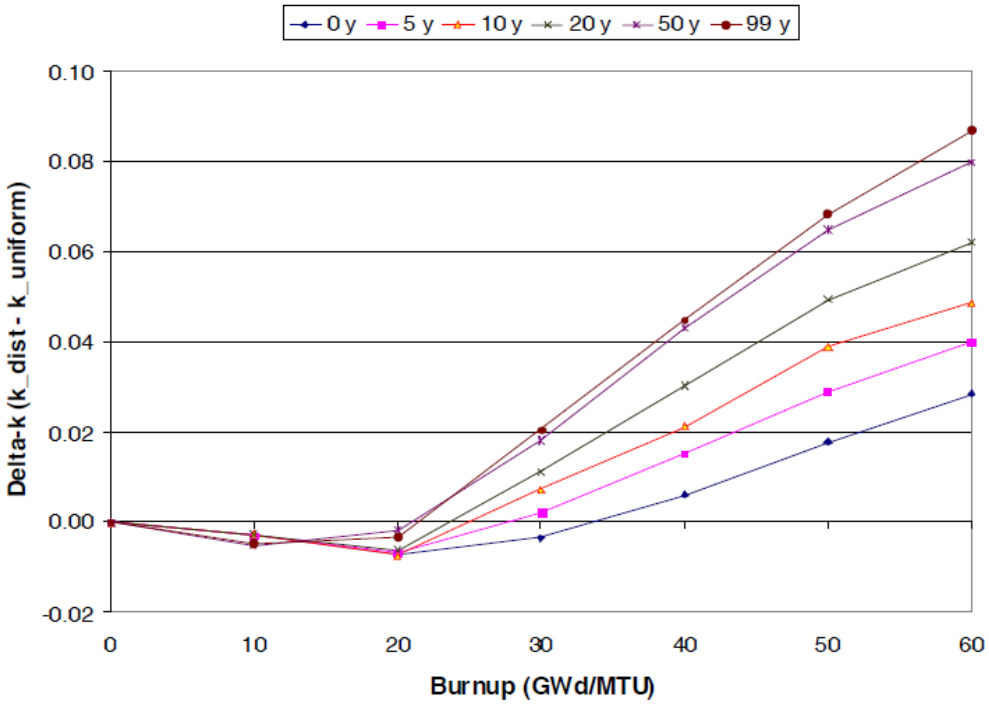

Figure 6A-7 Effect of axial burnup distribution on K_{eff} in the GBC-32 for actinide-only burnup credit and various cooling times for fuel with 4.0 weight percent initial enrichment (Source: Withee 2002)

reactive profiles, the nature of the database's limiting profiles, and their application to all assemblies in a storage cask or package, the database is adequate for obtaining bounding profiles for use in burnup credit analyses.

While the preceding discussion indicates that the database is an appropriate source of axial burnup profiles, the reviewer should ensure that profiles taken from the database are applied correctly. The application of the profiles in the database may not be appropriate for all assembly designs. This would include assemblies of different lengths than those evaluated in the database. While the database included some assemblies with axial blankets (natural or low enriched), these assemblies were not irradiated in a fully blanketed core (i.e., they were test assemblies). Thus, application of the database profiles to assemblies with axial blankets may also be inappropriate, as the impact of axial blankets has not been fully explored. However, it is generally conservative to assume that fuel is not blanketed, using the enrichment of the nonblanketed axial zone and the limiting axial profile.

Other sources of axial burnup profiles may be appropriate to replace or supplement the database of YAEC-1937. The reviewer should ensure that these other burnup profile sources are described and evaluated, similar to the treatment of the YAEC-1937 database in NUREG/CR-6801. The reviewer should ensure that the process used to obtain axial profiles included in the safety analysis has been described and that the profiles are justified as encompassing the realistic profiles for the entire burnup range over which they are applied. The process of selecting and justifying the appropriate bounding axial profile may be simplified and/or conservatism may be reduced if the axial burnup profile is measured before or during the package loading operation. The measurement should demonstrate that the actual assembly profile is equally or less reactive than that assumed in the safety evaluation.

Burnable Absorbers

Assemblies exposed to fixed neutron absorbers [also referred to as integral burnable absorbers (IBAs)] and removable neutron absorbers [also referred to as burnable poison rod assemblies (BPRs)] can have higher k_{eff} values than assemblies that are not exposed. This is because of the hardening of the neutron spectrum, and it will lead to increased fissile plutonium nuclide production and reduced uranium-235 depletion. In addition, when removable neutron absorbers are inserted, the spectrum is further hardened because of the displacement of the moderator. NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," issued March 2002, and NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit," issued March 2002, characterize the effects of burnable absorbers on SNF. The results of these studies indicate that a depletion analysis with a maximum realistic loading of BPRs (i.e., maximum neutron poison loading) and maximum realistic burnup for the exposure should provide an adequate bounding safety basis for fuel with or without BPRs. An evaluation relying on exposures to less than the maximum BPR loading or for less than the maximum burnup (for which credit is requested), or both, needs adequate justification for the selected values (e.g., provision of available data to support the value selection and/or indication of how administrative controls will prevent a misload of an assembly with higher exposure).

For IBAs, these studies indicate that the impact on k_{eff} depends on the material type and the burnup level. Exposure to the maximum absorber loading was seen to be bounding for zirconium diboride-type IBAs (known as integral fuel burnable absorbers) at burnups above about 30 GWd/MTU. At lower burnups, neglecting the presence of the absorber was seen to be bounding. Neglecting the absorber in the case of IBAs that use erbia, gadolinia, and alumina-boron carbide was also bounding for all burnups investigated for these IBAs. Exposures to absorber types or materials not considered in the references supporting this appendix, whether fixed, removable, or a combination of the two, should be evaluated on a case-by-case basis.

Control Rods

As with BPRs, CRs fully or partially inserted during reactor operation can harden the spectrum near the insertion and lead to increased production of fissile plutonium nuclides. In addition, CRs can alter the axial burnup profile. In either case, the CR would have to be inserted for a significant fraction of the total irradiation time for these effects to be seen in terms of a positive Δk on the SNF package. Domestic PWRs typically do not operate with CRs inserted, although the tips of the rods may rest right at the fuel ends. However, some older domestic reactors and certain foreign reactors may have used CRs more extensively, such that the impact of CR insertion would be significant.

Based on the results of NUREG/CR-6759, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit," issued February 2002, and the fact that BPRs and CRs cannot be inserted in an assembly at the same time, the inclusion of BPRs in the assembly irradiation model should adequately account for the potential increase in k_{eff} that may occur for typical SNF exposures to CRs during irradiation. However, inclusion of BPRs in the irradiation model may not fully account for exposures to atypical CR insertions (e.g., full insertion for one full reactor operation cycle), and assemblies irradiated under such operational conditions should be explicitly evaluated. Also, since the previously discussed axial burnup profile database (NUREG/CR-6800) includes a representative sampling of assemblies exposed to CRs and APSRs, the appropriate selection of a limiting axial profile(s) from that database would be

expected to adequately encompass the potential impact for axial profile distortion caused by CRs and APSRs.

Exposures to CR or APSR insertions or materials not considered in the references supporting this attachment should be explicitly evaluated. This would also apply to exposures to flux suppressors (e.g., hafnium suppressor inserts) or similar hardware that affects reactivity. Safety analyses for exposures to these items should use assumptions (e.g., duration of exposure, cycle(s) of exposure) that provide an adequate bounding safety basis and include appropriate justification for those assumptions. Additionally, the axial burnup and power distributions in assemblies exposed to these devices may be unusual; thus, it may be necessary to use actual axial burnup shapes for those assemblies.

Depletion Analysis Computational Model

For depletion analyses, computer codes that can track a large number of nuclides should be used to estimate the SNF nuclide concentration. Although certain nuclides that are typically tracked may not directly affect the concentrations of the nuclides in Tables 6A-1 and 6A-2, they can indirectly impact the production and depletion via their effect on the neutron spectrum. An accurate depletion analysis model requires tracking of a sufficient number of nuclides, use of accurate nuclear data, and prediction of burnup-dependent cross sections representative of the spatial region of interest.

Two-dimensional codes are routinely used together with axial segmentation of the fuel assembly in the criticality model to approximate axial variation in depletion. The two-dimensional flux calculations can capture the planar neutron flux distribution in each axial segment of a fuel assembly. The two-dimensional model is built to calculate the isotopic composition of the assembly at a series of burnup values, derived from the chosen axial burnup profile and the assembly-average burnup. This approach is acceptable because it accounts for both the planar and axial flux variation to achieve a relatively accurate depletion simulation. Ideally, three-dimensional computer codes would be useful for fuel assembly depletion analyses to accurately simulate this phenomenon. However, three-dimensional depletion analysis codes are not recommended at this time because of their current limitations.

Several two-dimensional codes based on neutron transport theory are available, such as CASMO, HELIOS, and the SCALE TRITON sequence (DeHart 2009). The reviewer should be aware of the limitations of a particular code and version, such as those designed to use lumped cross sections for multiple nuclides. Such limitations may require additional justification of the code's utility for burnup credit criticality analyses. Review of depletion analyses should focus on the suitability and accuracy of the code and modeling of the fuel assembly depletion history.

Previously, because of the limited availability of accurate two-dimensional computer codes, most burnup credit calculations used one-dimensional depletion codes to determine SNF isotopic concentrations averaged over the assembly. With appropriate code benchmarking against assay measurements and appropriate treatment of the fuel assembly spatial heterogeneity [e.g., Dancoff factor correction, disadvantage factor correction (Duderstadt and Hamilton 1976)], one-dimensional physics models of PWR assembly designs can produce sufficiently accurate assembly-average SNF compositions. However, to use a one-dimensional model, a cylindrical flux-weighted and geometry-equivalent supercell depletion model needs to be constructed to preserve the effective fuel assembly neutronics characteristics. Burnup-dependent cross sections are then generated using the flux-weighted and geometry-modified point-depletion model. This approach is sensitive to the accurate construction of the supercell materials and the approximation of the assembly geometry.

It is essential that the burnup-dependent cross sections are updated with sufficient frequency in the depletion analysis model and that the physics model used to update the cross sections is representative of the assembly design and reactor operating history. As with analyses used to determine k_{eff}, the depletion analysis should be appropriately validated. The application analysis should use the same code and cross-section library and the same, or similar, modeling options as were used in the depletion-validation analysis. Section 6A.5 of this attachment discusses in greater detail the issues associated with isotopic depletion-code validation.

Models for Prediction of k_{eff}

In addition to this SRP, the following documents address the expectations regarding the codes and modeling assumptions to be used to determine k_{eff} of an SNF transportation package:

- NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," issued April 1997

- NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," issued March 1997

Such applications typically require Monte Carlo codes capable of three-dimensional solutions of the neutron transport equation. A loading of SNF, including specific combinations of assembly-average burnup, initial enrichment, and cooling time, should be used for each package analysis. However, unlike unirradiated fuel, the variability of the burnup (and thus the isotopic concentrations) along the axial length is an important input assumption.

In particular, the burnup gradient will be large at the ends of the fuel regions. Thus, the package model should include several fuel zones, each with isotopic concentrations representative of the average burnup across the zone. Burnup profile information from reactor operations is typically limited to 18–24 uniform axial regions. NUREG/CR-6801 has shown that subdividing the zones beyond those provided in the profile information (assuming at least 18 uniform axial zones) yields insignificant changes in the k_{eff} value for a storage cask or package.

In reality, the end regions of the fuel have the lowest burnup and contribute the most to the reactivity of the system. Thus, the model boundary condition at the ends of the fuel will potentially be of greater importance than for uniform or fresh fuel cases where the reactivity in the center of the fuel dominates reactivity. The end-fitting regions above and below the fuel contain steel hardware with a significant quantity of void space (typically 50 percent or more) for potential water inleakage. The analyses in Appendix A to NUREG/CR-6801 demonstrate that modeling the end regions as either 100-percent steel or full-density water provides a higher value of k_{eff} than a combination (homogenized mixture 50-percent water and 50-percent steel assumed) of the two. For the storage cask or package that was studied, the all-steel reflector provided a k_{eff} change of nearly 1 percent over that of full-density water. Although use of 100-percent steel is an extreme boundary condition (since water will always be present to some degree), the results indicate that the applicant should take care to select a conservative boundary condition for the end regions of the fuel.

The large source of fissions distributed nonuniformly, because of the axial burnup profile, over a large source volume in an SNF package, can cause difficulty in properly converging the analysis to the correct k_{eff} value. Problems performed in an international code-comparison study (Blomquist et al. 2006) demonstrate that results can vary based on user selection of input parameters crucial to proper convergence. Strategies that may be used in the calculations to

accelerate the source convergence (e.g., starting particles preferentially at the more reactive end regions) should be justified and demonstrated to be effective.

An important issue in burnup credit criticality modeling is the need to verify that the correct SNF composition associated with the depletion and decay analysis is inserted in the correct spatial zone in the package model. The data-processing method to select and extract the desired nuclide concentrations from the depletion and decay analyses and input them correctly to the various spatial zones of the criticality analysis is not a trivial process and has the potential for error. The reviewer should verify the interface process, the computer code used to automate the data handling, or both. As with fresh fuel criticality analyses, the reviewer should verify that the criticality analyses for burnup credit are appropriately validated. In other words, the application analysis should use the same code and cross-section library and the same, or similar, modeling options as were used in the criticality-code validation. Section 6A.6 of this attachment discusses in greater detail the issues associated with criticality-code validation.

6A.5 Code Validation—Isotopic Depletion (Section 6.4.7.3 of this SRP)

An isotopic-depletion code typically consists of three parts:

1. a library of nuclear reaction cross sections

2. a geometric and material representation of the fuel assembly as well as the reactor core configuration

3. an algorithm to predict the isotopic transmutation over time as the fuel assembly is irradiated in the reactor and decays after discharge

To ensure the accuracy of the code and identify the biases and uncertainties associated with the algorithm, nuclear data, and modeling capability, the depletion code should be validated against measured data from RCA measurements of SNF samples.

Validation of the depletion-analysis code serves two purposes. The first purpose is to determine if the code is capable of accurately modeling the depletion environment of fuel assemblies for which burnup credit is taken. The second is to quantify the bias and bias uncertainty of the depletion code against the depletion parameters, fuel assembly design characteristics, initial enrichment, and cooling time.

In general, validation of the depletion code consists of the following steps:

1. Select RCA sample data sets that are suitable for validation of the depletion code.

2. Build and run depletion models for SNF samples that are selected for depletion-code validation.

3. Apply the bias and bias uncertainty of the depletion calculation to the criticality-analysis code implicitly through the use of adjusted isotopic concentrations of the depletion model, or determine the bias and bias uncertainties associated with the fuel-depletion-analysis code in terms of Δk_{eff}, as discussed in NUREG/CR-7108.

Selection of Validation Data

Validation data consist of measurements of isotopic concentrations from destructive RCA samples of SNF. Reliable depletion-code validation results require a sufficient number of data sets that include all isotopes for which burnup credit is taken. The applicant, therefore, should provide justification of the sample size for each nuclide. For example, the applicant should demonstrate that isotopic uncertainty is appropriately increased to account for uncertainty associated with limited available measurement data or for uncertainty associated with nonnormal isotopic validation data. The analyses in NUREG/CR-7108 use appropriate methods to account for these uncertainties.

Sample data necessary for depletion-code validation include initial enrichment and burnup, depletion history, assembly design characteristics, and physical location within the assembly. Over the past several decades, different laboratories have performed various RCA measurements of SNF samples. The NRC and ORNL have published detailed descriptions and analyses of the RCA measurements available for use in isotopic-depletion validation in the following references:

- NUREG/CR-7012, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," issued January 2011

- NUREG/CR-7013, "Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor I," issued January 2011

- NUREG/CR-6968, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors," issued February 2010

- NUREG/CR-6969, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—ARIANE and REBUS Programs (UO_2 Fuel)," issued February 2010

NUREG/CR-7108 analyzes the available data sets and identifies 100 fuel samples suitable for depletion-code validation for SNF storage and transportation systems. The reviewer should examine the sample data and depletion models to ensure that these sample data are used in the application to determine the bias and bias uncertainty associated with the chosen isotopic-depletion methodology. If different RCA data are used for the isotopic-depletion validation, the applicant should provide all relevant information associated with that data (e.g., burnup, enrichment, cool time, local irradiation environment), and justify that these data are appropriate for the intended purpose. RCA data from samples with incomplete or unknown physical and irradiation history data should be avoided. Note that the burnup values associated with the RCA measurements are the actual sample burnup rather than fuel assembly-average burnup, which is typically used in burnup credit calculations. Reviewers should ensure that the benchmark models the applicant constructed for depletion-code validation use the appropriate burnup value.

Because of differences in the techniques used in RCA measurement programs, in some cases, the results may vary significantly between different measurements of the same nuclide. These variations may result in a large uncertainty in the calculated concentration for a particular nuclide, and reviewers should expect to see such large uncertainties for certain nuclides until a better database of measurements is available.

Radiochemical Assay Modeling

The depletion-validation analysis should use the time-dependent irradiation environment and decay time for each individual RCA sample. Accurate sample depletion parameters should be used in the depletion-code validation analysis models. A sample should not be used if its depletion history and environment are not well known. Some samples were taken from specific locations in the fuel assembly, while other samples have been taken on an assembly-average basis. The latter type is typically found in earlier RCA data.

A depletion model should be built for each set of measurement data obtained from an RCA sample. To validate the computer code and obtain the bias and bias uncertainty, the depletion model should be able to accurately represent the environment in which each SNF sample was irradiated. For example, a sample from a fuel rod near a water hole will have a different neutron flux spectrum than a sample in a location where it is surrounded by fuel rods. Similarly, a fuel assembly with BPR insertion will have a different neutron spectrum in comparison to one without BPR exposure. Furthermore, a sample taken from the end of a fuel rod would have different specific power, fuel temperature, moderator temperature, and moderator density compared to those of a sample taken from the middle of a fuel assembly. Finally, time-dependent, three-dimensional effects, such as CR insertion, BPR insertions, and partial rod or gray rod insertions during part of the depletion processes, should also be captured. These local effects are typically averaged in a one-dimensional depletion code, and the reviewer should expect to see relatively large uncertainties associated with one-dimensional depletion-code calculations of individual RCA sample nuclide concentrations if this methodology is utilized.

Depletion-Code Validation Methods

One of the objectives of code validation is to determine the bias and bias uncertainty associated with the isotopic-concentration calculations. NUREG/CR-6811, “Strategies for Application of Isotopic Uncertainties in Burnup Credit,” issued June 2003, discusses several approaches to treating the bias and bias uncertainty associated with isotopic-concentration calculations. NUREG/CR-7108 expands on two of these approaches in greater detail and provides reference results for representative SNF storage and transportation systems. The following paragraphs discuss these approaches.

Isotopic Correction Factor Method

This approach uses a set of correction factors for isotopes that are included in burnup credit analyses. Correction factors are derived by statistical analysis of the ratios of the calculated-to-measured isotopic concentrations of the RCA samples for each isotope. The mean value, plus or minus the standard deviation multiplied by a tolerance factor appropriate to yield a 95/95 confidence level, is determined as the correction factor for a specific isotope. For the fissile isotopes, the correction factor is the mean value plus the modified standard deviation. For nonfissile absorber isotopes, the correction factor is the mean value minus the modified standard deviation. Fissile isotope correction factors that are below 1.0 should be conservatively set to 1.0, and absorber isotope correction factors that are above 1.0 should be conservatively set to 1.0. Since this method includes all the uncertainties associated with the measurements, computer algorithm, data library, and modeling, and since the correction factors are modified only in a manner that will increase k_{eff}, the result is considered bounding.

Direct-Difference Method

The direct-difference method directly computes the k_{eff} bias and bias uncertainty associated with the depletion code for the same set of isotopes by using the measured and calculated isotopic concentrations in the criticality analysis models separately. Two k_{eff} values are obtained in each pair of calculations, and a Δk_{eff} is calculated for each set of measured data. A statistical analysis is performed to calculate the mean value and the uncertainty associated with the mean value of the Δk_{eff}. Regression analysis is performed to determine the bias of the mean Δk_{eff} value as a function of various system parameters (e.g., burnup, initial enrichment).

The direct-difference method requires a full set of measured data for all isotopes for which this method is used to determine the bias and bias uncertainty of the isotopic-depletion analysis code. However, many isotopes in Tables 6A-1 and 6A-2, particularly the fission products, do not have sufficient measured data to allow significant statistical analysis. In these cases, surrogate data have been used, as described in NUREG/CR-7108. This surrogate data set was generated using the available measured data for an isotope as the basis for populating the missing data in the measured data sets. A surrogate data value was determined by multiplying the calculated nuclide concentration by the mean value of the measured-to-calculated concentration ratio values obtained from samples with measured data. The fundamental assumption of this approach is that the limited available measured data are representative of the entire population of isotopic concentration values. When the available measured data for a specific isotope are limited or cover a small burnup range, the applicant should ensure that this assumption is still valid, as Section 6.2 of NUREG/CR-7108 did for molybdenum-95, ruthenium-101, rhodium-103, and cesium-133.

Based on the studies published in NUREG/CR-7108, decay time correction is an important factor when using the direct-difference method. In cases where the cooling times of the samples used in code validation differ from the design-basis fuel cooling time, the error in the isotopic calculations can be large. NUREG/CR-7108 discusses the method for correcting decay times for the samples selected for code validation. This method uses the Bateman Equation (Benedict et al. 1981) to adjust the measured isotopic concentration of the nuclide of interest to the design-basis cooling time of the application. For a general case of nuclide B with a decay precursor A and a daughter product C (i.e., A → B → C), the content of nuclide B at a reference cooling time can be obtained by solving the Bateman Equation. The time-adjusted isotopic concentration should be used in the validation rather than the measurement data. In the case where only a fraction of the decay leads to the production of nuclide B, the fraction of decay of nuclide A leading to nuclide B should also be included. For a nuclide without a significant precursor, the contribution from decay of precursors should be set to zero, and only the decay of nuclide B need be considered.

Monte Carlo Uncertainty Sampling Method

The Monte Carlo uncertainty sampling method generates a depletion code k_{eff} bias (β_i) and bias uncertainty (Δk_i,) for the group of nuclides for which burnup credit is taken. It determines the bias and bias uncertainty using a statistical method that adjusts the isotopic concentrations of the SNF in the criticality analysis model by a factor randomly sampled within the uncertainty band of measured-to-calculated isotopic concentration ratios of each nuclide. NUREG/CR-7108 discusses this approach in more detail. Research results published in NUREG/CR-7108 indicate that this method, although statistically complex and computationally intensive, can be used to determine a more realistic bias and bias uncertainty of the depletion code.

Using the Monte Carlo uncertainty sampling method, ORNL has developed reference bias and bias uncertainty values for the hypothetical GBC-32 storage and transportation system. The NRC finds it acceptable for the applicant to directly use the bias and bias uncertainty values from Tables 6A-3 and 6A-4, in lieu of an explicit depletion validation analysis, provided that the following conditions are met:

- The applicant uses the same depletion code and cross-section library as used in NUREG/CR-7108 (SCALE/TRITON and the ENDF/B-V or ENDF/B-VII cross-section library).

- The applicant can justify that its design is similar to the hypothetical GBC-32 system design used as the basis for the NUREG/CR-7108 isotopic-depletion validation.

- Credit is limited to the specific nuclides listed in Tables 6A-1 and 6A-2 of this attachment.

Bias values should be added to the calculated system k_{eff}, while bias uncertainty values may be statistically combined with other independent uncertainties, consistent with standard criticality safety practice. Demonstration of package similarity to the GBC-32 should consist of a comparison of materials and geometry, as well as neutronic characteristics such as H/X ratio, EALF, neutron spectra, and neutron reaction rates. If any of the above conditions is not met, the applicant should use the direct-difference or isotopic-correction factor methods discussed previously.

6A.6 Code Validation—k_{eff} Determination (Section 6.4.7.4 of this SRP)

For the k_{eff} component of burnup credit criticality calculations, validation is the process by which a criticality code system user demonstrates that the code and associated data predict actual system k_{eff} accurately. The criticality code validation process should include an estimate of the bias and bias uncertainty associated with using the codes and data for a particular application.

American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," states the following:

> Bias shall be established by correlating the results of critical and exponential experiments with results obtained for these same systems by the calculational method being validated.

The previous technical basis for burnup credit in ISG-8, Revision 2, limited credit to the major actinides, since there were not adequate critical experiments at the time for estimating the bias and bias uncertainty relative to modeling SNF in a storage cask, or package, environment. This technical basis considered the fact that no critical experiments existed that included the fission-product isotopes important to burnup credit. Additionally, critical experiments available for actinide validation were limited to only (i) fresh low-enriched UO_2 systems and (ii) fresh mixed uranium and plutonium oxide [mixed oxide (MOX)] systems. These systems are not entirely representative of SNF in a transportation package, as fresh UO_2 systems contain no plutonium, and the MOX experiments generally do not have plutonium isotopic ratios consistent with those of burned fuel.

Table 6A-3 Isotopic k_{eff} bias uncertainty (Δk_i) for the representative PWR SNF system model using ENDF/B-VII data (β_i = 0) as a function of assembly-average burnup		
Burnup (BU) Range (GWd/MTU)	**Actinides Only Δk_i**	**Actinides and Fission Products Δk_i**
0≤BU<5	0.0145	0.0150
5≤BU<10	0.0143	0.0148
10≤BU<18	0.0150	0.0157
18≤BU<25	0.0150	0.0154
25≤BU<30	0.0154	0.0161
30≤BU<40	0.0170	0.0163
40≤BU<45	0.0192	0.0205
45≤BU<50	0.0192	0.0219
50≤BU≤60	0.0260	0.0300

Table 6A-4 Isotopic k_{eff} bias (β_i) and bias uncertainty (Δk_i) for the representative PWR SNF system model using ENDF/B-V data as a function of assembly-average burnup		
Burnup (BU) Range (GWd/MTU)[a]	**β_i for Actinides and Fission Products**	**Δk_i for Actinides and Fission Products**
0≤BU<10	0.0001	0.0135
10≤BU<25	0.0029	0.0139
25≤BU≤40	0.0040	0.0165
[a]Bias and bias uncertainties associated with ENDF/B-V data were calculated for a maximum of 40 GWd/MTU. For higher burnups, applicants should provide an explicit depletion-code validation analysis using one of the methods described in this attachment, along with appropriate RCA data.		

While there were no representative critical experiments for SNF transportation or storage criticality validation, there were RCA data that were considered adequate for validating actinide isotopic-depletion calculations for major actinide absorbers. For this reason, as well as the criticality-validation limitations discussed above, the NRC staff deemed it appropriate to recommend "actinide-only" credit for SNF transportation and storage criticality-safety evaluations. This approach represented the bulk of the reduction in k_{eff} resulting from depletion of the fuel (see Table 6A-5) and excluded the fission products, which served as additional margin to cover uncertainties from modeling of actinide depletion k_{eff} effects.

Although there continue to be insufficient critical experiments for a traditional validation of the code-predicted reduction in k_{eff} resulting from fission products and minor actinides in SNF, a group of critical experiments designed for validating SNF k_{eff} reduction resulting from major actinides has become available since ISG-8, Revision 2, was published. NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," issued September 2008, describes these actinide criticality validation data in detail. The data are available to applicants from ORNL, subject to execution of a nondisclosure agreement. These experiments are more appropriate for validating the code-predicted reduction in k_{eff} resulting from actinide depletion than are the fresh UO_2 or other MOX critical experiments. The HTC experiments consisted of fuel pins fabricated from mixed uranium and plutonium oxide, with the uranium and plutonium isotopic ratios designed to approximate what would be expected from UO_2 fuel burned in a PWR to 37.5 GWd/MTU. While these experiments were designed to correspond to a single burnup rather than the range of burnups that would be ideal for criticality validation, this data set represents a significant improvement to the criticality validation data available for actinide isotopes.

Table 6A-5 Fission product reactivity worth for "typical" burnup in generic burnup credit cask (GBC-32) with 4 weight percent uranium-235 Westinghouse 17×17 OFA, burned to 40 GWd/MTU			
Credited Nuclides	**k_{eff}**	**Δk**	**%Δk[a]**
Fresh Fuel	1.13653		
8 Major Actinides[b]	0.94507	0.19146	71.9
All Actinides	0.93486	0.01021	3.8
Key 6 Fission Products[c]	0.88499	0.04987	18.7
All Remaining Fission Products	0.87010	0.01489	5.6
Totals		0.26643	100

[a]This is the percentage of total Δk for the burnup attributable to the portion of the total nuclide population in the first column.
[b]Eight major actinides include uranium-235, uranium-238, plutonium-238, plutonium-239, plutonium-240, plutonium-241, plutonium-242, and americium-241.
[c]Six key fission products include rhodium-103, cesium-133, samarium-149, samarium-151, neodymium-143, and gadolinium-155.

The improvement to the actinide criticality validation data set allows applicants for burnup credit in SNF transportation packages and storage casks to perform a traditional validation of the actinide component of the reduction in k_{eff} resulting from burnup, following the recommendations of NUREG/CR-6361. NUREG/CR-7109 contains ORNL's representative actinide criticality validation for the GBC-32 transportation and storage system using the best available validation data.

Although the contribution from fission products to the reduction in k_{eff} resulting from burnup is relatively small (see Table 6A-5), applicants for SNF transportation packages have requested the additional credit represented by these absorbers. The apparent need for fission product credit results from the significant increase in the percentage of discharged PWR fuel assemblies that can be stored or shipped in a high-capacity (e.g., 32-assembly) system. Figure 6A-8 represents a typical discharged PWR fuel population in terms of initial enrichment and burnup. Two representative loading curves, one for actinide-only burnup credit and another for actinide and fission product burnup credit, are overlain on this figure, showing the relative amounts of the PWR fuel population that would be transportable in a hypothetical package. Although the loading curve does not move significantly from actinide-only credit to actinide and fission product credit, the curve moves across the bulk of the discharged fuel population, making a greater percentage of this population transportable. If more transportation packages have this high capacity, then the total number of eventual SNF shipments could be reduced.

The ability to properly validate criticality codes for actinide burnup credit is a crucial step toward recommending fission product credit, as the actinides represent the bulk of the reduction in k_{eff} resulting from burnup. However, it is still necessary to be able to estimate the bias and bias uncertainty that result from modeling fission products in SNF. Even so, critical experiments that include fission product absorbers continue to be exceedingly rare. As of this writing, there are only a handful of such publicly available critical experiments: one set involving samarium-149 (LEU-COMP-THERM-050), another involving rhodium-103 (LEU-COMP-THERM-079), and a third involving elemental samarium, cesium, rhodium, and europium (LEU-MISC-THERM-005).[6] The preferred method for further fission product criticality validation would be the development of numerous and varied critical experiments involving both actinide and fission product absorbers

[6] The Nuclear Energy Agency's "International Handbook of Evaluated Criticality Safety Benchmark Experiments," which is updated and published annually, describes these three sets of experiments.

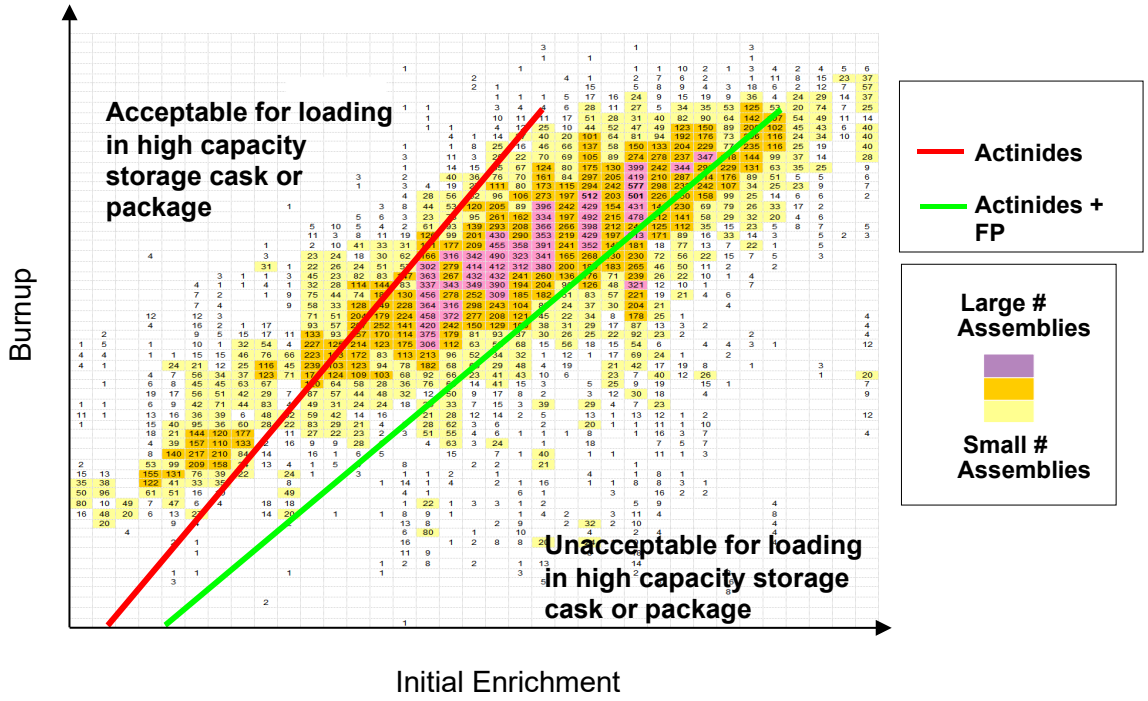

Figure 6A-8 Representative loading curves and discharged PWR population

in concentrations representative of SNF of various initial enrichments and burnups. Given the cost and practical difficulties associated with such a critical experiment program (e.g., obtaining specific absorber isotopes as opposed to natural distributions of isotopes), the NRC staff does not expect to see such experiments carried out within a reasonable timeframe. In the absence of such important criticality validation data, the NRC staff and contractors at ORNL sought alternative methods for estimating fission product bias and bias uncertainty.

To achieve an appropriate estimate of the k_{eff} bias and bias uncertainty for fission products, ORNL developed a methodology based on the SCALE Tools for Sensitivity and Uncertainty Methodology Implementation (TSUNAMI) code (Rearden 2009), developed as part of the SCALE code system. This methodology uses the nuclear data uncertainty estimated for each fission product cross section known as the "cross section covariance data." These data are provided with the ENDF/B-VII cross section library. The TSUNAMI code is used to propagate the cross section uncertainties represented by the covariance data into k_{eff} uncertainties for each fission product isotope used in a particular application. The theoretical basis of this validation technique is that computational biases are primarily caused by errors in the cross section data, which are quantified and bounded, with a 1σ confidence, by the cross section covariance data. NUREG/CR-7109 discusses the validity of this theoretical basis in greater detail.

This methodology has been benchmarked against the large number of low-enrichment uranium critical experiments, high-enrichment uranium critical experiments, plutonium critical experiments, and mixed uranium and plutonium critical experiments to demonstrate that the k_{eff} uncertainty estimates the method generated are consistent with the calculated biases for these systems. The k_{eff} uncertainty results for specific fission products were also compared to fission product bias estimates obtained from the limited number of critical experiments that include

fission products. NUREG/CR-7109 describes the uncertainty analysis method and provides details of the comparisons. The results demonstrate that, for a generic SNF transportation package evaluated with the SCALE code system and the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross-section libraries, the total fission product nuclear data uncertainty (1σ) does not exceed 1.5 percent of the total minor actinide and fission product worth for the 19 nuclides (Table 6A-2) considered over the burnup range of interest (i.e., 5 to 60 GWd/MTU). Since the uncertainty in k_{eff} resulting from the uncertainty in the cross-section data is an indication of how large the actual code bias could be, the 1.5-percent value should be used as a bias (i.e., it should be added directly to the calculated k_{eff}). Because of the conservatism in this value, no additional uncertainty in the bias needs to be applied.

To use the 1.5-percent value directly as a bias, applicants must demonstrate that they have used the code in a manner consistent with the modeling options and initial assumptions used in NUREG/CR-7109. Applicants must also demonstrate that their SNF transportation package design is similar to the GBC-32 used to develop the bias estimate. This demonstration should consist of a comparison of materials and geometry, as well as neutronic characteristics such as H/X ratio and EALF. Since improved actinide validation with the HTC experiments discussed previously represents a considerable part of the technical basis for crediting fission product absorbers, applicants should validate the actinide portion of the k_{eff} evaluation against this data set.

Applicants may also use a different criticality code if the code uses ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross-section data. In this case, the combined minor actinide and fission product bias and bias uncertainty should be increased to 3.0 percent. NUREG/CR-7109 shows that the bias and bias uncertainty are based largely on the uncertainty in the nuclear data. However, there are differences in how different codes handle the same cross-section data, potentially affecting bias and bias uncertainty. Since validation studies similar to that performed in NUREG/CR-7109 have not been performed for other codes, the staff finds that an additional k_{eff} penalty should be applied to cover any other uncertainties, and that doubling the 1.5 percent determined for the SCALE code system is conservative. ORNL performed additional analyses with MCNP5 and MCNP6, with ENDF/B-V, ENDF/B-VI, ENDF/B-VII, and ENDF/B-VII.1 cross-section data. These analyses, documented in NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks,” issued September 2015, demonstrate that the 1.5-percent value is also acceptable for use with these codes and cross-section libraries.

The reviewer should consider applicant requests to use the 1.5-percent value for other well-qualified industry standard code systems, provided that the application includes justification that this value is appropriate for that specific code system (e.g., a minor actinide and fission product worth comparison to SCALE results). For applications in which the applicant uses cross section libraries other than ENDF/B-V, ENDF/B-VI, or ENDF/B-VII, the transportation package cannot be demonstrated to be similar to the GBC-32, or the credited minor actinide and fission product worth is significantly greater than 0.1 in k_{eff}, an explicit validation analysis should be performed to determine the bias and bias uncertainty associated with minor actinides and fission products.

Integral Validation

ANSI/ANS 8.27-2008, “Burnup Credit for LWR Fuel,” provides a burnup credit criticality validation option consisting of analysis of applicable critical systems consisting of irradiated fuel with a known irradiation history. This is known as integral, or “combined,” validation, since the

bias and bias uncertainty associated with the depletion calculation method is inseparable from that associated with the criticality calculation method. The most common publicly available sources of integral validation data are commercial reactor critical (CRC) state points. These CRC state points consist of either a hot zero-power critical condition attained after sufficient cooling time to allow the fission product xenon inventory to decay or at-power equilibrium critical condition where xenon worth has reached a fairly stable value.

NUREG/CR-6951, "Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit," issued January 2008, shows CRC state points to be similar to storage cask-like and package-like environments, with respect to neutron behavior. With integral validation, however, the biases and uncertainties for the depletion approach cannot be separated from those associated with the criticality calculation, and only the net biases and uncertainties from the entire procedure are obtained. This approach allows for compensating errors between the depletion methodology and the criticality methodology (e.g., underprediction of a given nuclide's concentration coupled with simultaneous overprediction of this nuclide's effect on k_{eff}). It is desirable to understand the sources of uncertainty associated with the depletion methodology separately from those associated with the criticality methodology, to ensure that the overall bias and bias uncertainty are determined correctly for the transportation package, including package arrays, for the entire range of parameters.

Additionally, concerns remain about the physical differences between CRC state points and storage casks and transportation packages. These differences include borated water in a reactor versus fresh water in a package, high-worth absorber plates in a package versus none in a reactor, low moderator density in a reactor versus full density in a package, and high temperature in a reactor versus low temperature in a package. CRC state points also consist of calculated isotopic concentrations, as opposed to the measured concentrations one would expect in a typical laboratory critical experiment. Furthermore, CRC state points are inherently complicated to model, given the large number of assemblies and axial zones with different initial enrichments and burnups necessary to accurately model the reactor core. All of these concerns introduce additional uncertainties into a validation approach that attempts to use CRC state points.

For the reasons stated above, the staff does not recommend using integral validation approaches, with CRC state points or any other available integral validation data, for burnup credit criticality validation. However, if integral validation is used, the applicant should account for additional uncertainties, such as those identified above, and consider the use of a k_{eff} penalty to offset those uncertainties.

Loading Curve and Burnup Verification (Section 6.4.7.5 of this SRP)

As part of storage and transportation operations, loading curves are used to display acceptable combinations of assembly-average burnup and initial enrichment for loading fuel assemblies. Assemblies with insufficient burnup, in comparison with the loading curve, are not acceptable for loading, as shown in Figure 6A-8. Misloads have occurred in both dry storage casks and SNF pools, in which fuel did not satisfy allowable parameters (e.g., burnup, cooling time, and enrichment). Misloads occur because of misidentification, mischaracterization, or misplacement of fuel assemblies. In some cases, misloads have resulted in unanalyzed loading configurations during storage of SNF. To date, the known dry storage cask misload events have not had significant implications for criticality safety.

For efficiency and economic purposes in power plant operations, extraction of maximum power output from a fuel assembly before discharging it from the reactor is desirable. However, some

fuel assemblies have been removed from the reactor before achieving their desired burnup because of fabrication or performance issues. Once discharged from the reactor, these fuel assemblies are stored in the SNF pool. Because the SNF pool may contain assemblies with varying burnups, enrichments, and cooling times, a more reactive assembly could potentially be misloaded. Assemblies with fabrication issues, errors in reactor records, or operator actions that impact fuel handling activities are some of the several factors that can result in a misload.

ISG-8, Revision 3, specifies that certain administrative procedures should be established to ensure that fuel designated for a particular storage or transportation system is within the specifications for approved contents. The guidance recommends burnup measurement as a way to protect against misloads by identifying potential errors in reactor records or misidentification of assemblies being loaded into the system. As part of the overall initiative to revise the recommendations for the staff review of burnup credit criticality, the potential effects of misloaded assemblies on system reactivity were investigated.

Misloading of unirradiated fuel assemblies is unlikely for several reasons. First, storage and transportation system loading typically occurs when unirradiated fuel is not present in the SNF pool. Second, SNF is noticeably different than unirradiated fuel (e.g., color, deformation), and visually identifiable. Finally, the economic incentive involved with new fuel assemblies, would make permanent misloads of unirradiated fuel assemblies in dry storage casks or transportation packages unlikely.

Although misloading of unirradiated fuel assemblies is considered to be unlikely, an assembly that has been irradiated to less than the target burnup value (i.e., the assembly is underburned) could conceivably be misloaded into an SNF storage cask or transportation package. Misloading of one or more underburned fuel assemblies could increase the overall system reactivity. The amount of reactivity increase depends on several factors, including the degree of burnup in comparison to the loading curve, the cooling time, and the location of the assembly within the system.

The NRC has received reports of events involving misloads occurring within SNF pools and dry storage casks. Most of these misloads occurred as a result of inadequate fuel-selection procedures or inaccurate parameter data (i.e., burnup, enrichment, cooling time). Using available misload data, the RES report, “Estimating the Probability of Misload in a Spent Fuel Cask,” issued June 2011 (NRC 2011), evaluated the likelihood of misloading fuel assemblies within an SNF transportation package. This report determined the probability of single- and multiple-assembly misloads for ranges of burnup values dependent on the available SNF pool inventory. RES determined that the overall probability of misloading a fuel assembly that does not meet the burnup credit loading curve is in the range of 10^{-2} to 10^{-3}, which is considered credible.

NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” issued January 2008, evaluated the effects of single and multiple misloaded assemblies on the reactivity in a storage or transportation system. This evaluation covered the misloading of unirradiated and underburned PWR fuel assemblies in a GBC-32 high-capacity storage and transportation system. The scope of this report included varying the degree to which misloaded assemblies were underburned to determine the change in reactivity when including actinide-only and actinide and fission product burnup credit. The analysis covered a range of enrichments up to 5.0 weight percent uranium-235, while placing between one and four misloaded assemblies into the most reactive positions within the system. All assemblies within the system were assumed to undergo a cooling period of 5 years. The study evaluated the misloaded

assemblies at 90, 80, 50, 25, 10, and 0 percent (unirradiated) of the minimum assembly-average burnup value required by the loading curve.

The evaluation in NUREG/CR-6955 concluded that for the particular system design and fuel assembly parameters used, a reactivity increase between 2.0 and 5.5 percent in k_{eff} could be expected for various misloaded systems. Given the operational history and the accuracy of the reactor records, this information can be used along with the misload probability to determine an appropriate method of addressing assembly misloads as part of the criticality evaluation. Applicants may perform a misload analysis in lieu of a confirmatory burnup measurement.

Misload Evaluation

The applicant's misload evaluation should be based on a reliable and relatively recent estimate of the discharged PWR fuel population and should reflect the segment of that population that is intended to be stored or transported in the storage cask or package design. This population may consist of the entire population of discharged PWR fuel assemblies; a specific design of PWR fuel assembly (e.g., W17x17 OFA); or a smaller, specific population from a particular site. As of this writing, the 2002 Energy Information Administration (EIA) RW-859, "Nuclear Fuel Survey" (EIA 2004), is an acceptable source of discharged fuel data, although more recent data may be available (i.e., GC-859, "Nuclear Fuel Survey" (EIA 2015).

An applicant's misload analysis should evaluate both a single, severely underburned misload and a misload of multiple moderately underburned assemblies in a single SNF storage cask or package. The single severely underburned assembly should be chosen such that any assembly-average burnup and initial enrichment along an equal reactivity curve bound 95 percent of the discharged fuel population considered unacceptable for loading in the applicant's storage cask or transportation package with 95-percent confidence. Applicants should provide a statistical analysis of the underburned fuel population to support the selection of severely underburned assemblies.

The 95/95 criterion for evaluations of single high-reactivity misloads, in combination with the administrative procedures for misload prevention (see Administrative Procedures below), is reasonably bounding as more reactive misloads are unlikely. The assembly-average burnup and initial enrichment that match this 95/95 criterion are dependent on the loading curve for the storage or transportation system. Applicants are likely to seek a level of burnup credit that results in qualification of the greatest possible amount of the fuel population for storage or shipment in the system. Therefore, assemblies matching the 95/95 criterion will be those with relatively high enrichment and low burnup (e.g., 5 weight percent uranium-235 and 15 GWd/MTU). Based on the data in the 2002 EIA RW-859, the number of discharged assemblies of greater reactivity is very small, even for cases where all discharged assemblies of a given burnup and initial enrichment are located in a single SNF pool.

For the evaluation of the applicant's storage cask or package with multiple moderately underburned assemblies, misloaded SNF should be assumed to make up at least 50 percent of the system payload and should be chosen such that the assembly-average burnups and initial enrichments along the equal reactivity curve bound 90 percent of the total discharged fuel population. Such an evaluation is reasonably bounding for cases of multiple misloads in a single SNF storage cask or package based on the considerations in the following paragraph.

The 90-percent criterion is based on the total discharged fuel population and not the specific loading curve for the system design. The distribution of discharged fuel peaks within a relatively narrow band of burnup for each initial enrichment value. The curve that represents a reactivity

that bounds 90 percent of the discharged population is expected to pass through burnup and enrichment combinations that are below this peak. However, the population along this curve is still large enough to represent possible misload scenarios involving multiple assemblies. Below the 90-percent criterion curve, with few exceptions, the numbers of assemblies for each burnup and enrichment combination drop significantly. Thus, it is reasonable to expect that misloading of multiple assemblies of the remaining 10 percent of the discharged population would be less likely. Although there are larger numbers of low-burnup assemblies for specific initial enrichments, facilities that have a significant number of these assemblies can reduce the likelihood of misloading multiples of these assemblies in the same storage cask or package with proper administrative controls.

The recommendation for assuming misloading of at least 50 percent of the system is based on consideration of the history of misloads in dry SNF storage operations and the fact that systematic errors can result in misloading of multiple assemblies. Misloads that have occurred in dry SNF storage operations have typically involved multiple assemblies. The most significant of these incidents resulted in less than 25 percent of the storage cask capacity being misloaded. While the probability of a multiple-misload scenario decreases with increasing number of assemblies involved, systematic errors can increase the likelihood of such misloads. Considering these factors, there is reasonable assurance that a scenario that involves misloading at least 50 percent of the storage cask or package capacity would bound the extent of likely multiple-misload conditions. The implementation of the administrative procedures recommended in Section 6.4.7.5 of this SRP and in this attachment for preventing misloads provides additional assurance against more extensive misload situations.

It is possible that SNF storage casks and packages designed for specific parts of the fuel population (e.g., particular sites or fuel types) will have loading curves that already bound 90 percent of the discharged fuel population. In these cases, misload analysis for multiple assemblies is not necessary.

An SNF storage or transportation system should be designed to have a limited sensitivity to misloads, such that increases in k_{eff} when considering misloads are minimized. In any case, the applicant should demonstrate that the system remains subcritical under misload conditions, including biases, uncertainties, and an administrative margin. As in the nominal loading analyses, the misload analyses should use the design parameters and specifications that maximize system reactivity. The administrative margin is normally 0.05. However, for misload evaluations, a different administrative margin may be used, given two conditions. First, the administrative margin should not be less than 0.02. Second, any use of an administrative margin less than 0.05 should be adequately justified. An adequate justification should consider the level of conservatism in the depletion and criticality calculations, sensitivity of the system to further upset conditions, and the level of rigor in the code-validation methods.

An administrative margin is used with criticality evaluations to ensure that a system that is calculated to be subcritical is actually subcritical. This margin is used to ensure against unknown errors or uncertainties in the method of calculating k_{eff}, as well as impacts of system design and operating conditions not explicitly considered in the analysis. Criticality safety practices in other regulated areas give allowance for using different administrative margins. Experience with identified code errors and an understanding of uncertainties in cross-section data and their impacts on reactivity indicate that an administrative margin of at least 0.02 is necessary for analyses to show subcriticality with misloads.

Taking credit for burnup reduces the margin in the analyses and makes them more realistic. Additionally, decreasing the administrative margin for misload analyses further reduces the margin for subcriticality. This reduction in overall criticality safety margin necessitates greater justification for a lower administrative margin. The justification should demonstrate a greater level of assurance that the various sources of bias and bias uncertainty have been considered and that the bias and bias uncertainty are known to a high degree of accuracy. The principles and concepts discussed in Division of Fuel Cycle Safety and Safeguards ISG-10, "Justification for Minimum Margin of Subcriticality for Safety" (NRC 2000), are useful in understanding the kinds of evaluations and evaluation rigor that should be considered for justification of a lower administrative margin. These concepts include assurances of the consistent presence and degree of conservatism in the evaluations that may be relied on, the quality and number of benchmark experiments as they relate to the application and the misload cases, and evaluation of the sensitivity of k_{eff} to other system parameter changes.

Administrative Procedures

Along with the misload analysis, administrative procedures should be established in addition to those procedures typically performed for non-burnup credit systems. The purpose of these additional procedures is to ensure that the system will be loaded with fuel that is within approved technical specifications or CoC conditions. Procedures considered to protect against misloads in storage and transportation systems that rely on burnup credit for criticality safety may include the following:

- verification of the location of high-reactivity fuel (i.e., fresh or severely underburned fuel) in the SNF pool both before and after loading
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement)
- under an NRC-approved quality assurance program, verification before shipment of the inventory and loading records of a canister or storage cask that was previously loaded and placed into dry storage and that is to be shipped in or as the package
- quantitative measurement of any fuel assemblies without visible identification numbers
- independent, third-party verification of the loading process, including the fuel selection process and fuel move instructions
- (for dry storage under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste") minimum soluble boron concentration in pool water, to offset the misloads described above, during loading and unloading

Most of these recommendations are intended to ensure that high-reactivity fuel is not present in the pool during loading or is otherwise accounted for and determined not to have been loaded into an SNF storage system or transportation package. The verification of the storage system inventory and loading records before loading and shipment in a package is intended to ensure that the contents of previously loaded storage systems are as expected before shipment. This verification should be performed under an approved 10 CFR Part 71, "Packaging and Transportation of Radioactive Materials" quality assurance program.

Quantitative measurement of SNF without visible identification is recommended since there is no other apparent way to demonstrate that such assemblies are tied to a specific burnup value.

Independent, third-party verification of the fuel selection process means verification of the correct application of fuel-acceptability standards and the fuel move instructions.

Soluble boron is recommended as an unloading condition to ensure that misloads are protected against when future unloading operations occur, since the conditions of such operations are currently unknown and may inadvertently introduce unborated water into the system. Soluble boron is typically present during PWR SNF loading operations for dry storage or transportation systems. An appropriate soluble boron concentration during loading and unloading would be that required to maintain system k_{eff} below 0.95 with the more limiting (in terms of k_{eff}) of the single, severely underburned or multiple moderately underburned misloads described previously. Consistent with requirements such as those in 10 CFR 71.55(b), transportation package analyses cannot credit the soluble boron present during PWR SNF loading into or unloading from the package. Therefore, the discussion regarding use of a minimum soluble boron concentration during loading and unloading (and credit for this soluble boron in analyses) applies only to loading and unloading for dry storage under 10 CFR Part 72.

This revision of the criticality safety review guidance for burnup credit in the SRP includes misload analyses as an alternative to burnup confirmation using measurement techniques. A number of misloads have occurred within SNF pools and storage casks as a result of human errors or inaccurate assembly data. Efforts have been made to evaluate the criticality effects of misloading assemblies into an SNF transportation package. Using credible bounding assumptions, a misload analysis could be generated to account for potential events during loading, while maintaining an appropriate safety margin.

6A.7 References

10 CFR Part 71, “Packaging and Transportation of Radioactive Materials.”

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007), “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,” American Nuclear Society, La Grange Park, IL.

ANSI/ANS 8.27-2008, “Burnup Credit for LWR Fuel,” American Nuclear Society, La Grange Park, IL.

Benedict, M., T.H. Pigford, and H.W. Levi, *Nuclear Chemical Engineering*, Second Edition, McGraw Hill, New York, NY, 1981.

DeHart, M.D., “Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages,” ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, Oak Ridge, TN, May 1996.

DeHart, M.D., “Triton: A Two-Dimensional Transport and Depletion Module for Characterization of Spent Nuclear Fuel,” ORNL/TM-2005/39, Version 6, Vol. I, Section T1, Oak Ridge, TN, January 2009.

DOE/RW-0496, “Horizontal Burnup Gradient Datafile for PWR Assemblies,” U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Washington, DC, May 1997.

Duderstadt, J.J. and L.J. Hamilton, *Nuclear Reactor Analysis*, John Wiley & Sons Inc., Hoboken, NJ, 1976.

Blomquist, Roger N., Malcolm Armishaw, David Hanlon, Nigel Smith, Yoshitaka Naito, Jinan Yang, Yoshinori Mioshi, Toshihiro Yamamoto, Olivier Jacquet, and Joachim Miss, “Source Convergence in Criticality Safety Analyses, Phase I: Results for Four Test Problems,” NEA No. 5431, NEA OECD, 2006.

MCNP5, “MCNP—A General Monte Carlo N-Particle Transport Code, Version 5; Volume II: User’s Guide,” LA-CP-03-0245, Los Alamos National Laboratory, Los Alamos, NM, April 2003.

NRC, ISG-8, Revision 2, “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks,” Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, DC, September 27, 2002.

NRC, ISG-8, Revision 3, “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks,” Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, DC, September 26, 2012.

NRC, ISG-10, Revision 0, “Justification for Minimum Margin of Subcriticality for Safety,” Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC, July 2000 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML061650370).

NRC, “Revisiting the Paradigm for Spent Fuel Storage and Transportation Regulatory Programs,” Staff Requirements Memorandum COMDEK-09-0001, U.S. Nuclear Regulatory Commission, Washington, DC, February 2010 (ADAMS Accession Nos. ML100491511 and ML100500024).

NRC, “Estimating the Probability of Misload in a Spent Fuel Cask,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, November 2011 (ADAMS Accession No. ML113191144).

Nuclear Energy Agency, “International Handbook of Evaluated Criticality Safety Benchmark Experiments,” Nuclear Science Committee, updated and published annually, https://www.oecd-nea.org/science/wpncs/icsbep/handbook.html.

NUREG/CR-5661, “Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages,” ORNL/TM-11936, Oak Ridge National Laboratory, Oak Ridge, TN, April 1997.

NUREG/CR-6361, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” ORNL/TM-13211, Oak Ridge National Laboratory, Oak Ridge, TN, March 1997.

NUREG/CR-6665, “Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel,” ORNL/TM-1999/303, Oak Ridge National Laboratory, Oak Ridge, TN, February 2000.

NUREG/CR-6747, “Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit,” ORNL/TM-2000/306, Oak Ridge National Laboratory, Oak Ridge, TN, October 2001.

NUREG/CR-6759, “Parametric Study of the Effect of Control Rods for PWR Burnup Credit,” U.S. Nuclear Regulatory Commission,” ORNL/TM 2001/69, Oak Ridge National Laboratory, Oak Ridge, TN, February 2002.

NUREG/CR-6760, “Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit,” ORNL/TM-2000/321, Oak Ridge National Laboratory, Oak Ridge, TN, March 2002.

NUREG/CR-6761, “Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit,” ORNL/TM-2000/373, Oak Ridge National Laboratory, Oak Ridge, TN, March 2002.

NUREG/CR-6781, “Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses,” ORNL/TM-2001/272, Oak Ridge National Laboratory, Oak Ridge, TN, January 2003.

NUREG/CR-6800, “Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs,” ORNL/TM-2002/6, Oak Ridge National Laboratory, Oak Ridge, TN, March 2003.

NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” ORNL/TM-2001/273, Oak Ridge National Laboratory, Oak Ridge, TN, March 2003.

NUREG/CR-6811, “Strategies for Application of Isotopic Uncertainties in Burnup Credit,” ORNL/TM-2001/257, Oak Ridge National Laboratory, Oak Ridge, TN, June 2003.

NUREG/CR-6951, “Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit,” ORNL/TM-2006/87, Oak Ridge National Laboratory, Oak Ridge, TN, January 2008.

NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” ORNL/TM-2004/52, Oak Ridge National Laboratory, Oak Ridge, TN, January 2008.

NUREG/CR-6968, “Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors,” ORNL/TM-2008/071, Oak Ridge National Laboratory, Oak Ridge, TN, February 2010.

NUREG/CR-6969, “Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-ARIANE and REBUS Programs (UO_2 Fuel),” ORNL/TM-2008/072, Oak Ridge National Laboratory, Oak Ridge, TN, February 2010.

NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data,” ORNL/TM-2007/083, Oak Ridge National Laboratory, Oak Ridge, TN, September 2008.

NUREG/CR-7012, “Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel,” ORNL/TM-2010/41, Oak Ridge National Laboratory, Oak Ridge, TN, January 2011.

NUREG/CR-7013, “Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor,” ORNL/TM-2009/321, Oak Ridge National Laboratory, Oak Ridge, TN, January 2011.

NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions,” ORNL/TM-2011/509, Oak Ridge National Laboratory, Oak Ridge, TN, April 2012.

NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions,” ORNL/TM-2011/514, Oak Ridge National Laboratory, Oak Ridge, TN, April 2012.

NUREG/CR-7194, “Technical Basis for Peak Reactivity Burnup Credit for BWR Spent Nuclear Fuel in Storage and Transportation Systems,” ORNL/TM-2014/240, Oak Ridge National Laboratory, Oak Ridge, TN, April 2015.

NUREG/CR-7205, “Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP K_{eff} Calculations for PWR Burnup Credit Casks,” ORNL/TM-2012/544, Oak Ridge National Laboratory, Oak Ridge, TN, September 2015.

Oak Ridge National Laboratory, “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, June 2011. Available as CCC-785 from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, Oak Ridge, TN, https://rsicc.ornl.gov/Catalog.aspx?c=CCC.

Rearden, B.T., “TSUNAMI-3D: Control Module for Three-Dimensional Cross-Section Sensitivity and Uncertainty Analysis for Criticality,” ORNL/TM-2005/39, Version 6, Vol. I, Section C9, Oak Ridge National Laboratory, Oak Ridge, TN, January 2009.

RW-859, “Nuclear Fuel Data Survey, U.S. Energy Information Administration, Washington, DC, https://www.eia.gov/nuclear/spent_fuel/.

Withee, C.J., Memorandum to M. Wayne Hodges, “ISG-8, Rev. 2, Supporting Document,” U.S. Nuclear Regulatory Commission, Washington, DC, September 27, 2002 (ADAMS Accession No. ML022700395).

YAEC-1937, “Axial Burnup Profile Database for Pressurized-Water Reactors,” Yankee Atomic Electric Company, May 1997. Available as Data Package DLC-201, PWR-AXBUPRO-SNL, from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, Oak Ridge, TN, https://rsicc.ornl.gov/Catalog.aspx?c=DLC.