APPENDIX 7A TECHNICAL RECOMMENDATIONS FOR THE CRITICALITY SAFETY REVIEW OF PRESSURIZED-WATER REACTOR TRANSPORTATION PACKAGES AND STORAGE CASKS THAT USE BURNUP CREDIT

7A.1 Introduction

The overall reactivity decrease of nuclear fuel irradiated in light water reactors is from 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 SNF reactivity caused by irradiation. Section 7.5.5 of this standard review plan (SRP) provides guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for its use in the review of SNF container designs that seek burnup credit. This appendix provides the technical bases for the burnup credit recommendations for dry storage provided in the SRP and for transportation; thus, the appendix discusses both storage and transportation. As noted in Section 7.5.5, these recommendations and their technical bases are based on dry storage system (DSS)-type storage container designs, which are commonly referred to as casks or storage systems in this appendix. Application of the recommendations to other SNF storage container designs (in specific license applications for dry storage facilities (DSFs)) should involve consideration of the differences between container designs that are like DSSs and those that are not and the applicability of the recommendations' technical bases to non-DSS-like containers. This is also true for application of the recommendations, in specific license applications, to criticality analyses for SNF in other relevant DSF SSCs (e.g., a pool that is part of the DSF design and operations).

Historically, criticality safety analyses for transportation and dry cask storage of SNF assumed the fuel contents to be unirradiated (i.e., "fresh" fuel). In 2002, the NRC Spent Fuel Project Office (SFPO) issued Interim Staff Guidance-8 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," Revision 2 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 published, 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 cask or package designs that credit burnup in the criticality analysis.

Since ISG-8, Revision 2, was published, 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 (SFST, formerly SFPO) by the NRC Office of Nuclear Regulatory Research (RES) and its contractors at Oak Ridge National Laboratory (ORNL). This appendix summarizes the findings of a number of reports and papers published as part of the research program directed by RES 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 appendix. A comprehensive bibliography of burnup credit-related technical reports and papers is provided at http://www.ornl.gov/sci/nsed/rnsd/pubs_burnup.shtml.

7A.2 General Approach in Safety Analysis

Criticality safety analyses of SNF storage or transportation systems involve a great deal of complexity in both the computer modeling of the system, as well as the necessary fuel information. The assumption of unirradiated fuel at maximum initial enrichment provides a straightforward approach for the criticality safety analysis of a SNF dry storage or transportation system. This approach is conservative in terms of criticality safety and limits the system capacity. In comparison to the fresh fuel assumption, performing 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. In order 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 down 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 a 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 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 7.5.5 of this SRP include the following:

- general information on limits for the licensing 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. An extension 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 provided 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 8.5.15.1 of this SRP provides guidance for classifying the condition of the fuel (e.g., damaged, intact) for SNF storage and transportation.

The validation methodologies presented in Sections 7A.5 and 7A.6 of this appendix were performed for a representative cask 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." As will be discussed later in this appendix, in order 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," and NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions," 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 provided in this review guidance were developed with PWR fuel as the basis. Boiling-water reactor (BWR) burnup credit has not typically been sought by dry storage and transportation applicants 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, 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 spent fuel 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,” 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 spent fuel assemblies at peak reactivity in storage or transportation systems. Consult NUREG/CR-7194 if the applicant uses peak reactivity BWR burnup credit methods in its criticality analysis.

Credit for BWR burnup beyond peak reactivity is not addressed in this SRP, and is currently being evaluated by an NRC research program to investigate methods for conservatively including such credit in a BWR criticality analysis for SNF storage systems. The NRC does not recommend burnup credit beyond peak reactivity at this time. Conservative analyses of BWR burnup credit beyond peak reactivity should be considered 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 appendix 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 safety analysis report (SAR).

7A.3 Limits for Licensing Basis (Chapter 7, Section 7.5.5.1 of the 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-reactor between 1 and 40 years.

7A.3.1 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.” This report concludes that the actinides and fission products listed in Tables 7A-1 and 7A-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 non-fissile absorbers) remain the same and have been substantiated by numerous 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 7A.5 and 7A.6 of this appendix discuss. Accurate prediction of the concentrations for the nuclides in Tables 7A-1 and 7A-2 requires that the depletion and decay calculations include nuclides beyond those listed in the tables. Additional actinides and fission products are needed to assure 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.

Table 7A-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 7A-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	

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 applicants 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.

7A.3.2 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 judgment 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 only be taken 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 with regard to the maximum initial enrichment that can be considered in a burnup credit analysis; thus, the initial enrichment limit for the licensing basis remains at 5.0 weight percent uranium-235.

7A.3.3 Cooling Time

Figure 7A-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}. The fact that reactivity begins to rise around 100 years after discharge 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 recently 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 a SNF transportation package or the CoC or license for dry storage may require an additional condition with regard to the applicability of the credited burnup of the SNF contents. The condition would be dependent upon the type of credit taken and the post irradiation decay time credited in the analysis. For example, crediting of 40 years would result in a CoC or license condition limiting the applicability of the credited burnup to 160 years after fuel discharge. Note that 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," provides a comprehensive study of the effect of cooling time on burnup credit for various cask designs and SNF compositions.

7A.3.4 Summary

The acceptance criteria for burnup credit are based on the characteristics of SNF discharged to date, the parameter ranges considered in the majority of 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 (1) demonstrate that the measurement or experimental data necessary for proper code validation have been included, and (2) provide adequate justification that the analysis assumptions or the associated bias and bias uncertainty have been established in such a fashion as to bound 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.

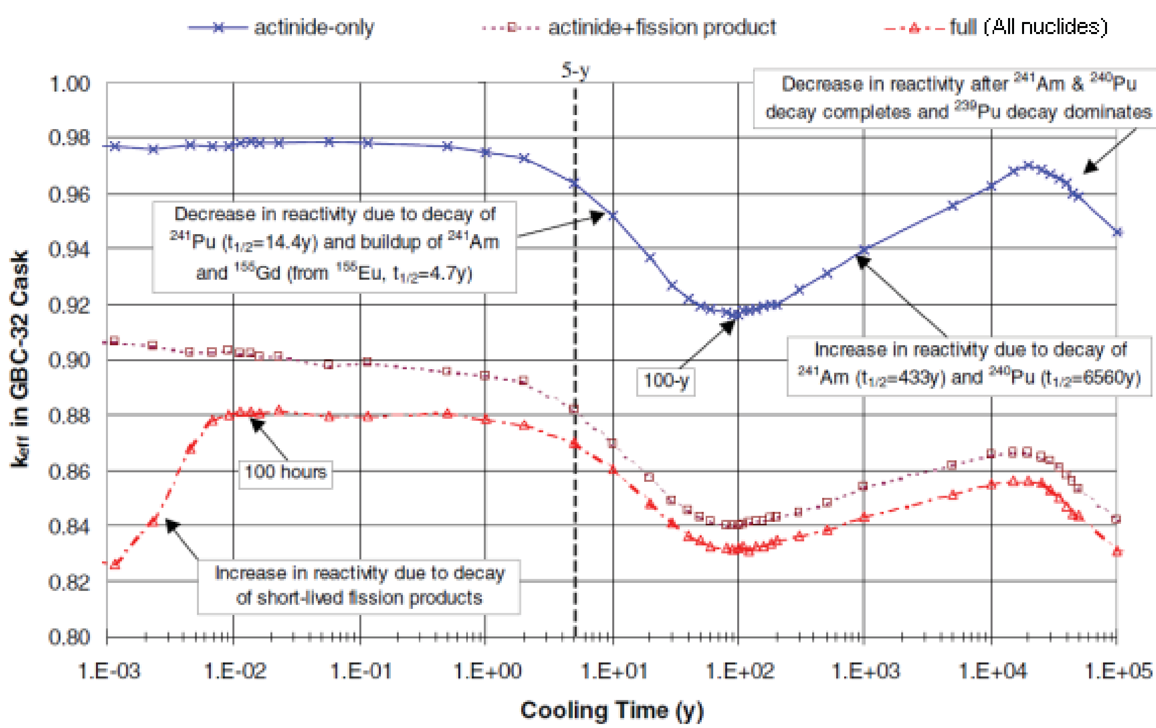

Figure 7A-1 Reactivity Behavior in The GBC 32 Cask as a Function of Cooling Time for Fuel with 4.0 Weight Percent Uranium-235 Initial Enrichment and 40 Gwd/MTU Burnup (Source: NRC 2010)

7A.4 Licensing-Basis Model Assumptions (Chapter 7, Section 7.5.5.2 of the SRP)

The actinide and fission product compositions used to determine a value of k_{eff} for the licensing basis should be calculated using fuel design and reactor operating parameter values that encompass the range of design and operating conditions for the proposed contents. Note that the proposed contents may consist of the entire population of discharged PWR fuel assemblies, a specific design of PWR fuel assembly (e.g., W17 × 17 optimized fuel assembly (OFA)), or a smaller, specific population from a particular site. The calculation of the k_{eff} value should be performed using cask models, analysis assumptions, and code inputs that allow accurate representation of the physics in the system. The following provides a discussion of important parameters that should be addressed in depletion analyses and k_{eff} calculations in a burnup credit evaluation.

7A.4.1 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 cask.

As the assumed fuel temperature used in the depletion model increases, the k_{eff} for the SNF in the cask 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 7A-2 to 7A-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 because of the parameter increase causing increased production of fissile plutonium nuclides and decreased uranium-235 utilization.

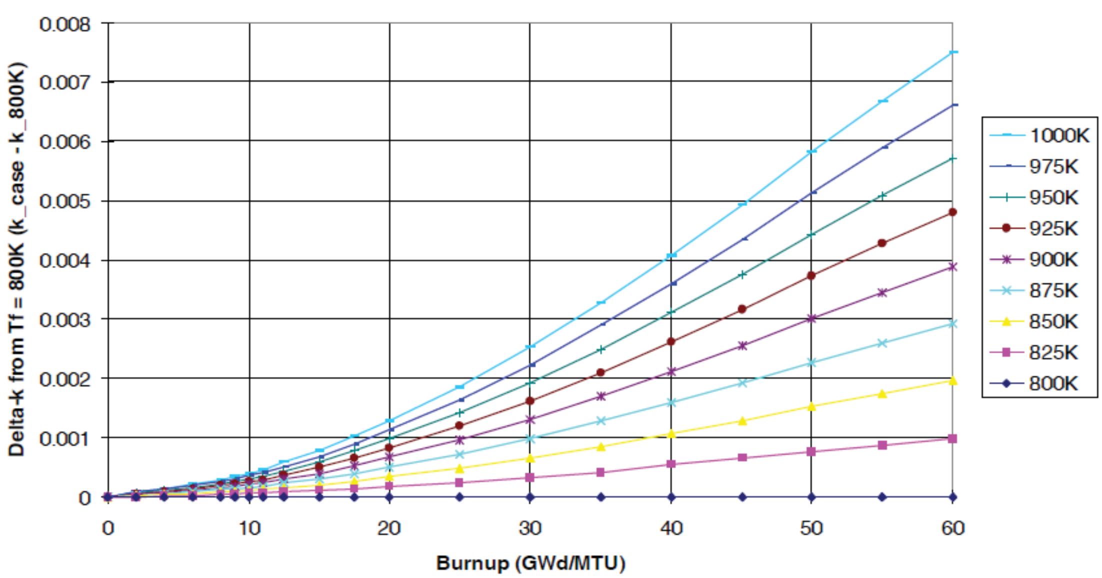

Figure 7A-2 Reactivity Effect of Fuel Temperature During Depletion on K_{inf} in an Array of Poisoned Storage Cells; Results Correspond to Fuel with 5.0 Weight Percent Initial Uranium-235 Enrichment (Source: Withee 2002)

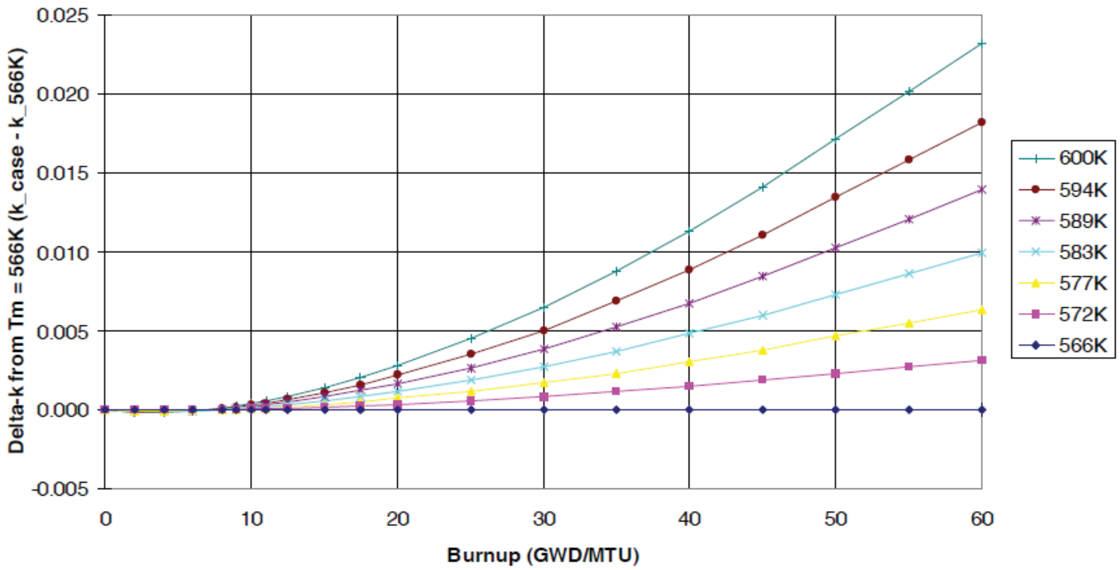

Figure 7A-3 Reactivity Effect of Moderator Temperature During Depletion on K_{inf} in an Array of Poisoned Storage Cells; Results Correspond to Fuel with 5.0 Weight Percent Initial Uranium-235 Enrichment (Source: Withee 2002)

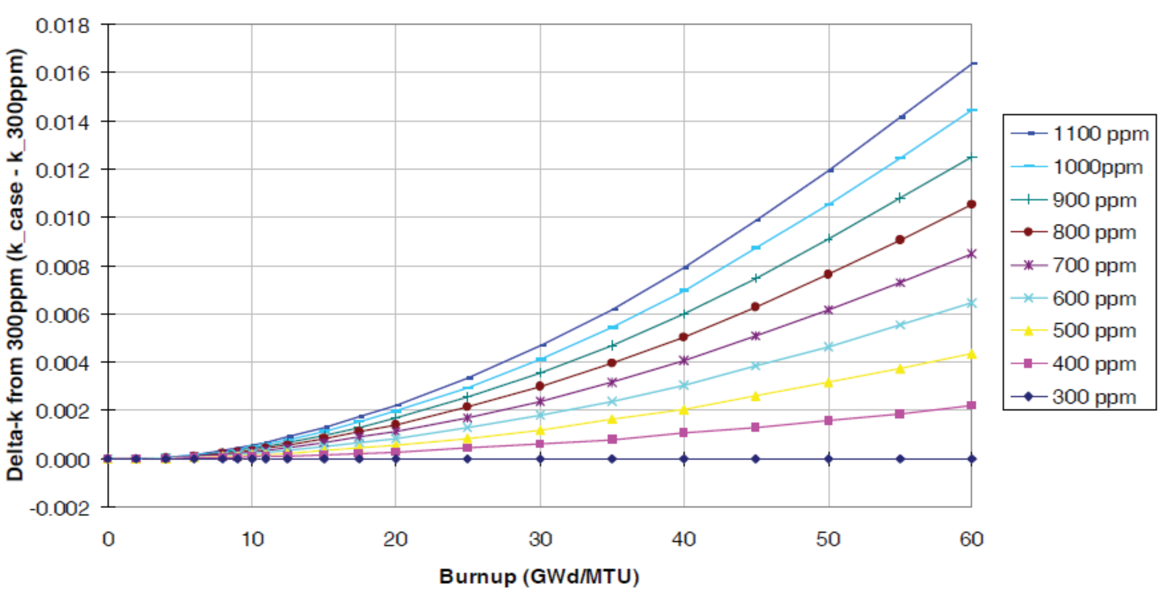

Figure 7A-4 Reactivity Effect of Soluble Boron Concentration During Depletion on K_{inf} in an Array of Poisoned Storage Cells; Results Correspond to Fuel with 5.0 Weight Percent Initial Uranium-235 Enrichment (Source: Withee 2002)

The impact of specific power and operating history is much more complex but has a very small impact on the cask k_{eff} value. Figures 7A-5 and 7A-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.

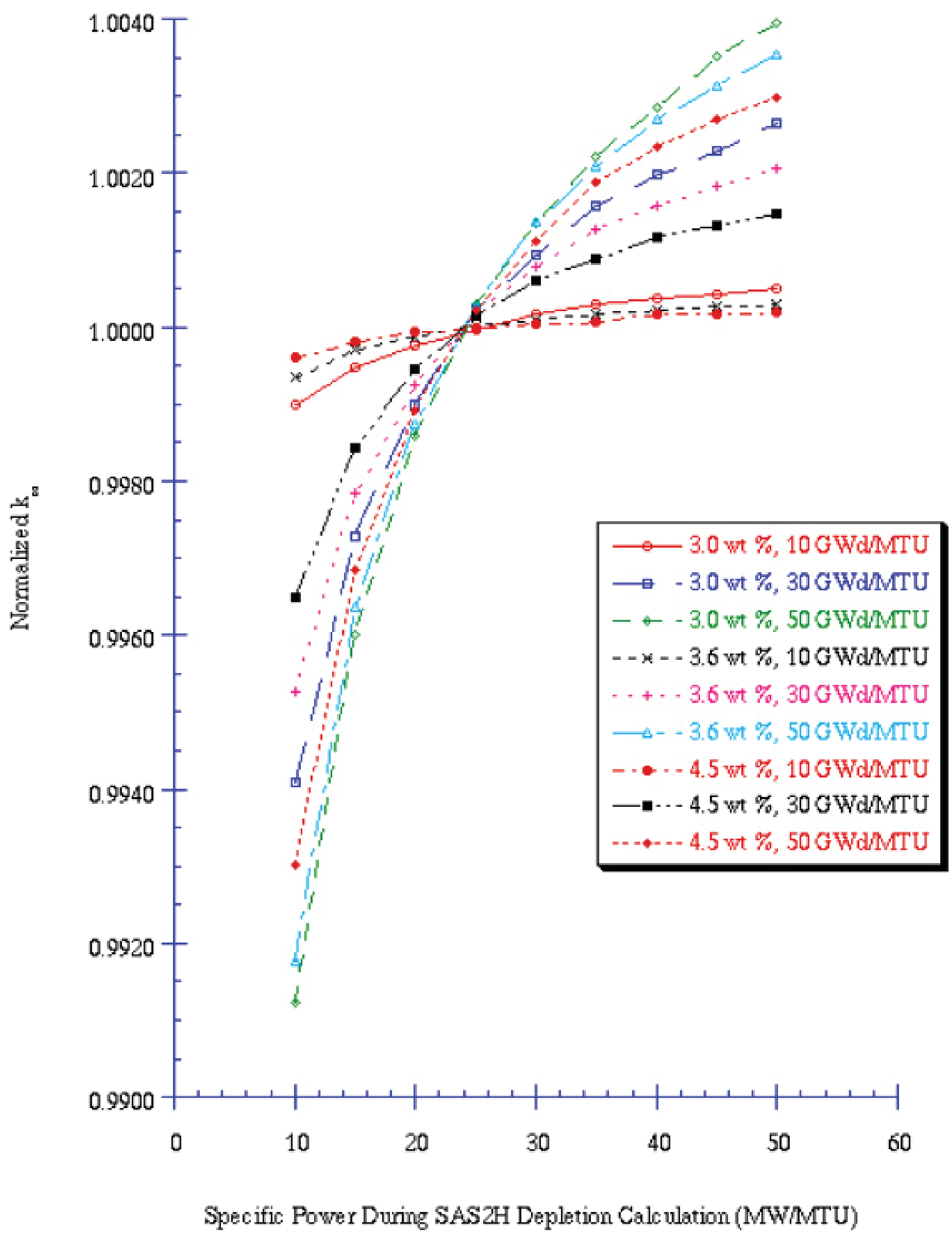

Figure 7A-5 Reactivity Effect of Specific Power During Depletion on k_{inf} in an Array of Fuel Pins (Actinides Only) (Source: DeHart 1996)

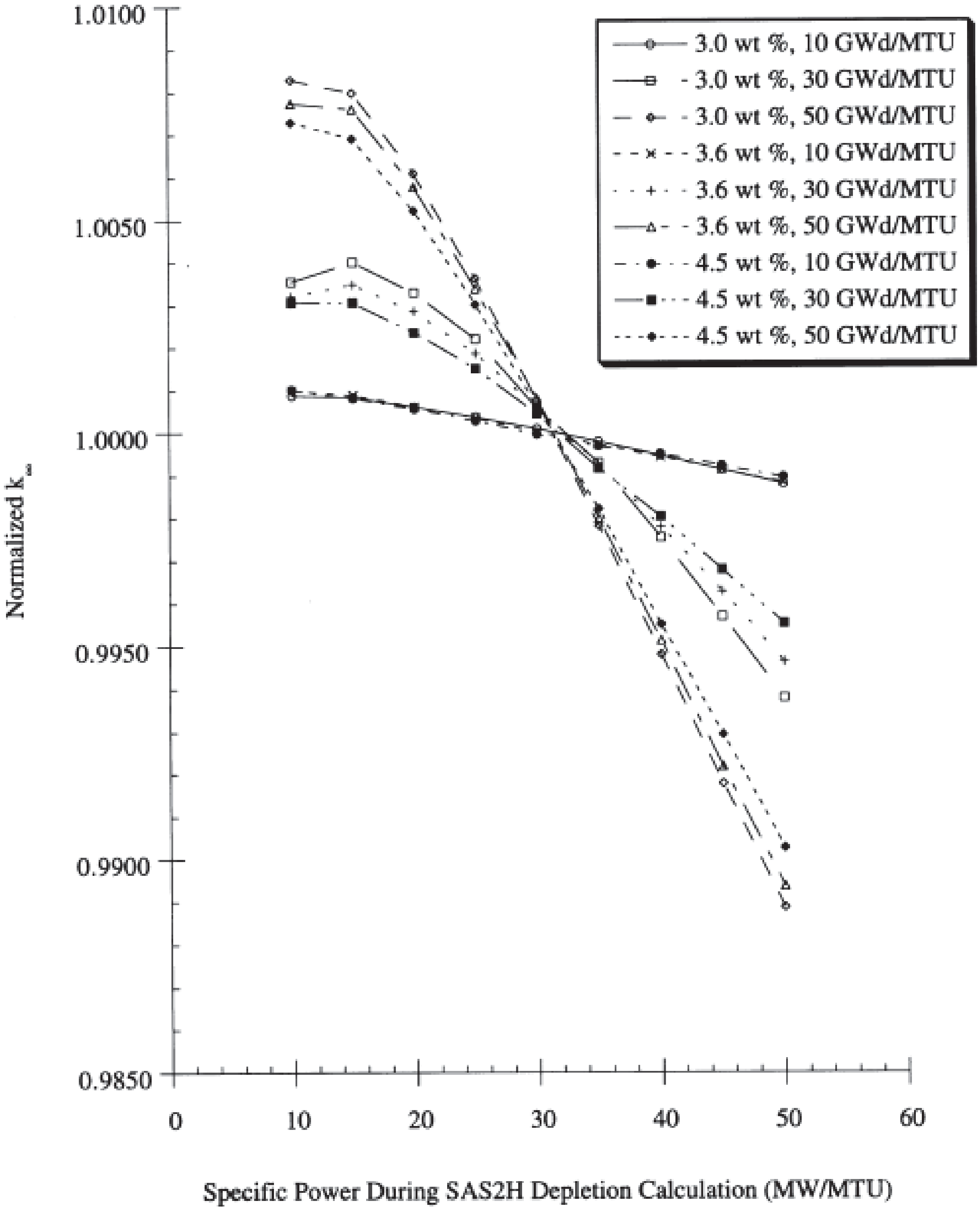

Figure 7A-6 Reactivity Effect of Specific Power During Depletion on K_{inf} in an Array of Fuel Pins (Actinides and Fission Products) (Source: DeHart 1996)

More detailed information on the impact of each parameter or phenomenon that should be assumed in the depletion model is provided in NUREG/CR-6665 and DeHart (1996). Each of the trends and impacts has been substantiated by independent studies. 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 cask. 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 inventory. 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. Note that 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").

Also, the most reactive fuel design prior to 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 cask 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 cask loading.

7A.4.2 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 cask. 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"). In large storage or rail casks, 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 system design or demonstrate that the assemblies to be loaded in the system will be verified to not have such gradients. One acceptable approach would be to determine the difference in k_{eff} for a system loaded with fuel having a horizontal burnup gradient and a system 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.

7A.4.3 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". Figure 7A-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 7.5.5.2 of this SRP and this appendix indicate that any analysis should provide “an accurate representation of the physics” in the system. 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 the fact that the axial effect will vary with burnup, cooling time, SNF nuclides used in the prediction of k_{eff}, and cask design. The reviewer should consider the range of profiles anticipated for the fuel to be loaded in the system.

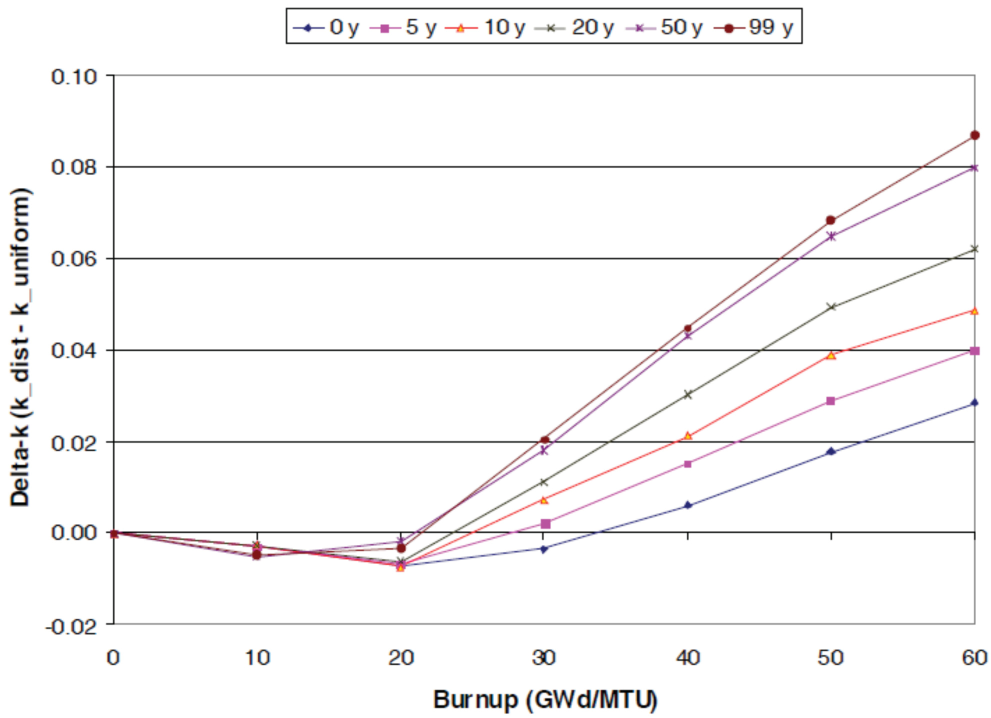

Figure 7A-7 Effect of Axial Burnup Distribution on K_{eff} in the GBC-32 Cask for Actinide-Only Burnup Credit and Various Cooling Times for Fuel with 4.0 Weight Percent Initial Enrichment (Source: Withee 2002)

The publicly available database of axial profiles in YAEC-1937, “Axial Burnup Profile Database for Pressurized Water Reactors,” is recommended as an appropriate source for selecting axial burnup profiles that will encompass the SNF anticipated for loading in a burnup credit cask. 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 per Section 7.5.5.1 of this SRP and Section 7A.3 of this appendix. 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 reactive profiles, the nature of the database's limiting profiles, and their application to all assemblies in a cask, 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 fuel is not blanketed, using the enrichment of the non-blanketed 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 assure that a description and evaluation of these other burnup profile sources similar to that demonstrated for the YAEC-1937 database in NUREG/CR-6801 has been performed. The reviewer should assure 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 a measurement of the axial burnup profile is performed before or during the cask loading operation. The measurement should demonstrate that the actual assembly profile is equally or less reactive than that assumed in the safety evaluation.

7A.4.4 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 due to the hardening of the neutron spectrum and 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," and NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit," provide characterizations of 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, the results of 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.

7A.4.5 Control Rods

As with BPRs, CRs fully or partially inserted during reactor operation can harden the spectrum in the vicinity of 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 cask. 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 in a more extensive fashion, 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, U.S. Nuclear Regulatory Commission," 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, exposures to atypical CR insertions (e.g., full insertion for one full reactor operation cycle) may not be fully accounted for by inclusion of BPRs in the irradiation model, 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 the guidance in Section 7.5.5 of this SRP and this appendix should be explicitly evaluated. This would also apply to exposures to flux suppressors (e.g., hafnium suppressor inserts) or similar hardware which affect 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.

7A.4.6 Depletion Analysis Computational Model

For depletion analyses, computer codes that can track a large number of nuclides should be used in order to obtain an accurate estimate of the SNF nuclide concentration. Although certain nuclides that are typically tracked may not directly impact the concentrations of the nuclides in Tables 7A-1 and 7A-2, they can indirectly impact the production and depletion via their effect on the neutron spectrum. Tracking of a sufficiently large number of nuclides, the use of accurate nuclear data, and the prediction of burnup-dependent cross sections representative of the spatial region of interest are necessary for an accurate depletion analysis model.

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 the present time because of their current limitations.

Several two-dimensional neutron transport theory based codes 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, in order 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 one that 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. Issues associated with isotopic depletion code validation are discussed in greater detail in Section 7A.5 of this appendix.

7A.4.7 Models for Prediction of k_{eff}

The expectations regarding the codes and modeling assumptions to be used to determine k_{eff} of a dry storage cask are documented in this SRP as well as the following documents:

- NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages"
- NUREG/CR-6361, Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages"

Monte Carlo codes capable of three-dimensional solutions of the neutron transport equation are typically required for such applications. A loading of SNF, including specific combinations of assembly-average burnup, initial enrichment, and cooling time, should be used for each cask 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 cask 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 that provided in the profile information (assuming at least 18 uniform axial zones) yields insignificant changes in the k_{eff} value for a cask.

In reality, the end regions of the fuel have the lowest burnup and provide the largest contribution 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 both 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 cask 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 be attentive to the selection of 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 a SNF cask, 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) have demonstrated 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 cask 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 that 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. Issues associated with criticality code validation are discussed in greater detail in Section 7A.6 of this appendix.

7A.5 Code Validation—Isotopic Depletion (Chapter 7, Section 7.5.5.3 of the 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 assure 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 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

7A.5.1 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 a small number of available measurement data or for uncertainty associated with non-normal isotopic validation data. The analyses in NUREG/CR-7108 use appropriate methods to account for these uncertainties.

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

- NUREG/CR-7012, “Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel”
- NUREG/CR-7013, “Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor”
- NUREG/CR-6968,” Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors”
- NUREG/CR-6969, “Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-ARIANE and REBUS Programs (UO_2 Fuel)”

NUREG/CR-7108 analyzes the available data sets and identified 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 this data is 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 constructed by the applicant for depletion code validation use the appropriate burnup value.

Because of differences in the techniques used in RCA measurement programs, the results may vary significantly between different measurements of the same nuclide, in some cases. 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.

7A.5.2 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. Note that 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 that were obtained from a 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 in comparison with that 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 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.

7A.5.3 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,” discusses several approaches to treat 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. These approaches are discussed in the following paragraphs.

1. 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 non-fissile absorber isotopes, the correction factor is the mean value minus the modified standard deviation. Fissile isotope correction factors that are below 1.0 are conservatively set to 1.0, and absorber isotope correction factors that are above 1.0 are 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 only modified in a manner that will increase k_{eff}, the result is considered bounding.

2. 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).

Note that 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 7A-1 and 7A-2, particularly the fission products, do not have sufficient numbers of measured data for performing 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 to populate 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 number of available measured data for a specific isotope is low or covers a small burnup range, the applicant should ensure that this assumption is still valid, as was done for molybdenum-95, ruthenium-101, rhodium-103, and cesium-133 in Section 6.2 of NUREG/CR-7108.

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 there are differences between the cooling times of the samples used in code validation and the design-basis fuel cooling time, the error in the isotopic calculations can be large. NUREG/CR-7108 provides a discussion of the method to correct decay times for the samples that were 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 needs to be considered.

3. 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 provides a more detailed discussion of this approach. 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 use the bias and bias uncertainty values from Tables 7A-3 and 7A-4 directly, in lieu of an explicit depletion validation analysis, provided the following conditions are met:

- the applicant uses the same depletion code and cross section library as was 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 7A-1 and 7A-2

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 system 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 are not met, the applicant should use the direct-difference or isotopic correction factor methods discussed previously.

Table 7A-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 7A-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 burnups higher than this, applicants should provide an explicit depletion code validation analysis using one of the methods described in this appendix, along with appropriate RCA data.

7A.6 Code Validation—K_{eff} Determination (Chapter 7, Section 7.5.5.4 of the 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.

As stated in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors":

> 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 cask environment. This technical basis considered the fact that no critical experiments existed which included the fission product isotopes important to burnup credit. Additionally, critical experiments available for actinide validation were limited to only (1) fresh low-enriched UO_2 systems and (2) fresh mixed uranium and plutonium oxide 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 that of burned fuel.

While there were no representative critical experiments for SNF transportation or storage criticality validation, there were considered to be adequate RCA data 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 that it was 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} due to depletion of the fuel (see Table 7A-5) and excluded the fission products that served as additional margin to cover uncertainties due to modeling actinide depletion k_{eff} effects.

Table 7A-5 FP Reactivity Worth for "Typical" Burnup in Generic Burnup Credit Cask (GBC-32) with 4 Weight Percent Uranium-235 Westinghouse 17 X 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 percent 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, 143Nd, and 155Gd.

Although there continue to be insufficient critical experiments for a traditional validation of the code-predicted reduction in k_{eff} due to fission products and minor actinides in SNF, a group of critical experiments designed for validating SNF k_{eff} reduction due to major actinides has become available since ISG-8, Revision 2, was published. This actinide criticality validation data is described in detail in NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," and is available to applicants from ORNL, subject to execution of a non-disclosure agreement. These experiments are more appropriate for validating the code-predicted reduction in k_{eff} resulting from actinide depletion than 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.

The improvement to the actinide criticality validation data set allows applicants for burnup credit SNF transportation packages and storage casks to perform a traditional validation for the actinide component of the reduction in k_{eff} resulting from burnup, per the recommendations of NUREG/CR-6361. ORNL has performed a representative actinide criticality validation for the GBC-32 transportation package provided in NUREG/CR-7109 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 7A-5), applicants for SNF transportation packages have requested the additional credit represented by these absorbers. The apparent need for fission product credit result from the significant increase in the percentage of discharged PWR fuel assemblies capable of being stored or shipped in a high-capacity (e.g., 32-assembly) system. Figure 7A-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 a greater number of transportation packages can have this high capacity, then the total number of eventual SNF shipments could be reduced.

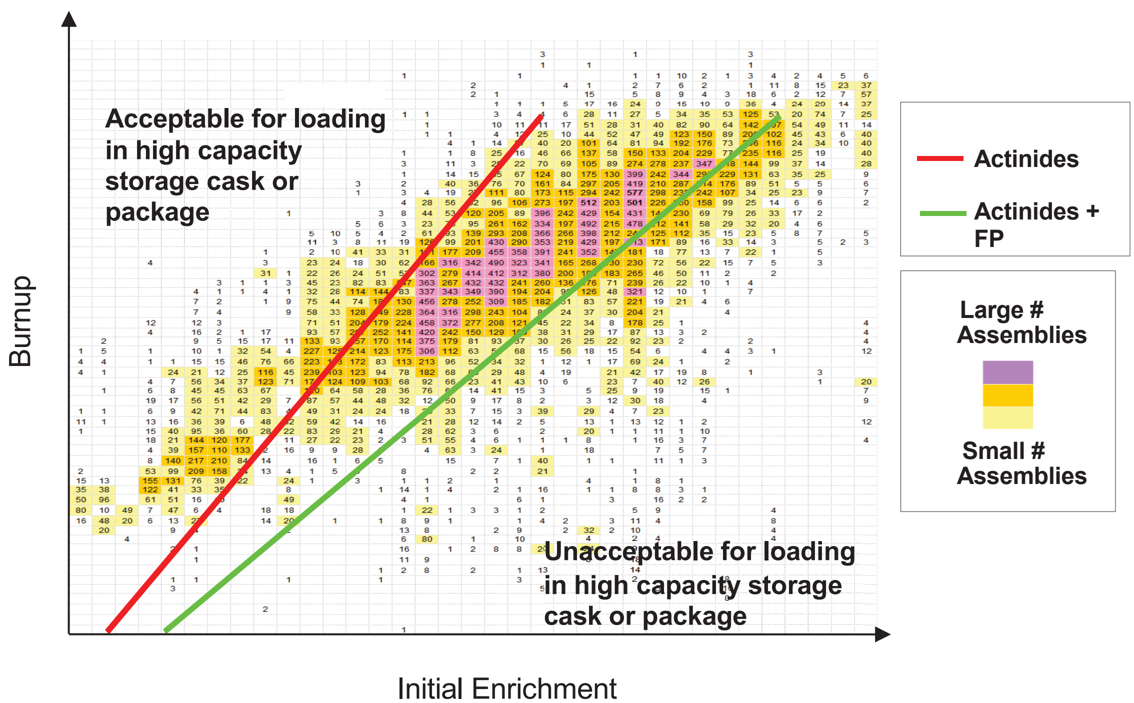

Figure 7A-8 Representative Loading Curves and Discharged PWR Population

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 due to modeling fission products in SNF, and 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). 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 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 methodologies for estimating fission product bias and bias uncertainty.

In order to achieve an appropriate estimate of the k_{eff} bias and bias uncertainty due to 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 a 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 generated by the method 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 7A-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., added directly to the calculated k_{eff}). Because of the conservatism in this value, no additional uncertainty in the bias needs to be applied.

In order 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 storage or transportation system 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, provided that 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 additional 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,” 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 the application includes additional 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 where the applicant uses cross section libraries other than ENDF/B-V, ENDF/B-VI, or ENDF/B–VII, where the application system cannot be demonstrated to be similar to the GBC-32, or where 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.

7A.6.1 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 an at-power equilibrium critical condition where xenon worth has reached a fairly stable value.

CRC state points have been shown to be similar to cask-like environments, with respect to neutron behavior, in NUREG/CR-6951, “Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit.” 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., under prediction of a given nuclide’s concentration coupled with simultaneous over prediction 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 for the criticality methodology in order to ensure that the overall bias and bias uncertainty are determined correctly for the cask system for the entire range of system parameters.

Additionally, concerns remain regarding the physical differences between CRC state points and cask systems, such as borated water in a reactor versus fresh water in a cask, high worth absorber plates in a cask versus none in a reactor, low moderator density in a reactor versus full density in a cask, and high temperature in a reactor versus low temperature in a cask. 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 make use of CRC state points.

For the reasons stated above, using integral validation approaches is not recommended, 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.

7A.7 Loading Curve and Burnup Verification (Chapter 7, Section 7.5.5.5 of the 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 7A-8. Misloads have occurred in both dry storage casks and spent fuel 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. This has resulted in unanalyzed loading configurations during storage of SNF in some cases. To date, the known dry storage cask misload events have not had significant implications on criticality safety.

For efficiency and economic purposes in power plant operations, it is desirable to ensure that the maximum power output is extracted from a fuel assembly before discharging it from the reactor. 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 spent fuel pool. Because the spent fuel pool may contain assemblies with varying burnups, enrichments, and cooling times, the potential for a more reactive assembly to be misloaded exists. A misload can occur as a result of several factors, including assemblies with fabrication issues, errors in reactor records, or operator actions which impact fuel handling activities.

ISG-8, Revision 2, specified 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. Burnup measurement was recommended in the guidance 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 staff burnup credit criticality review recommendations, 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 spent fuel pool. Second, SNF is noticeably different than unirradiated fuel (color, deformation) and visually identifiable. Finally, there is an economic incentive involved with new fuel assemblies that 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, it is conceivable that an assembly that has been irradiated to less than the target burnup value (i.e., underburned) could be misloaded into an SNF system. Misloading of one or more underburned fuel assemblies can cause an increase in 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.

A number of events involving misloads occurring within spent fuel pools and dry storage casks have been reported to the NRC. The majority 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,” (NRC 2011) evaluated the likelihood of misloading fuel assemblies within a SNF transportation package. This report determined the probability of single and multiple assembly misloads for ranges of burnup values dependent on the available spent fuel 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 10^{-2} to 10^{-3} range, which is considered credible.

NUREG/CR-6955, “Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask,” 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. This was done over a range of enrichments up to 5.0 weight percent uranium-235, while placing between 1 and 4 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 misloaded assemblies were evaluated 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.

7A.7.1 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 cask or package design. Note that this population may consist of the entire population of discharged PWR fuel assemblies, a specific design of PWR fuel assembly (e.g., W17 x 17 OFA), or a smaller, specific population from a particular site. An acceptable source of discharged fuel data as of this writing is the 2002 Energy Information Administration (EIA) RW-859, “Nuclear Fuel Data Survey” (EIA 2004), although more recent data may be available.

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 system. 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 a particular storage or transportation system 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, along 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 upon 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 of relatively high enrichment and low burnup (e.g., 5 wt. percent uranium-235 and 15 GWd/MTU). Based on the data available 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 spent fuel pool.

For the evaluation of the application system 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 system based upon 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 system with proper administrative controls.

The recommendation for misloading at least 50 percent of the system is based on consideration of the history of misloads in dry SNF storage operations as well as 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 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 cask capacity would bound the extent of likely multiple-misload conditions. The implementation of the administrative procedures recommended in Section 7.5.5.5 of this SRP and this appendix for preventing misloads provides additional assurance against more extensive misload situations.

It is possible that SNF systems 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, the misload analysis for multiple assemblies does not need to be performed.

A 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. The analysis should use the design parameters and specifications that maximize system reactivity as is done for nominal loading analyses. The administrative margin is normally 0.05. However, for the purposes of the 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 insure 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. Allowance for using different administrative margins is given in criticality safety practices in other regulated areas. Experience with identified code errors and an understanding of uncertainties in cross section data and their impacts on reactivity indicates 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, reducing the administrative margin for misload analyses further reduces the margin for subcriticality. This reduction in overall criticality safety margin necessitates a greater justification for a lower administrative margin. This justification should demonstrate a greater level of assurance that the various sources of bias and bias uncertainty have been taken into account and that the bias and bias uncertainty are known with a high degree of accuracy. The principles and concepts discussed in FCSS 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 amount of conservatism in the evaluations which may be relied upon, 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.

7A.7.2 Administrative Procedures

Along with the misload analysis, administrative procedures should be established, in addition to those typically performed for non-burnup credit systems, to ensure that the system will be loaded with fuel that is within approved technical specifications. 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 spent fuel pool both before and after loading

- qualitative verification that the assembly to be loaded is burned (visual or gross measurement)

- verification, under a 10 CFR Part 71 quality assurance program, of the system inventory and loading records before shipment for previously loaded systems

- 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

- use and confirmation of minimum soluble boron concentration in pool water, to offset the misloads described above, during loading and unloading of storage containers under 10 CFR Part 72

The majority 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 a SNF storage or transportation system. The verification of the system inventory and loading records is intended to ensure that the contents of previously loaded systems are as expected before shipment. This verification should be performed under an approved 10 CFR Part 71 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 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.

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

7A.8 References

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