Division of Spent Fuel Storage and Transportation
Interim Staff Guidance - 8
Revision 3

Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks

Introduction:

Title 10 of the Code of Federal Regulations (10 CFR) Part 71, *Packaging and Transportation of Radioactive Material*,[1] and 10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*,[2] require that spent nuclear fuel (SNF) remain subcritical in transportation and storage, respectively. Unirradiated reactor fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transportation and storage systems. As the fuel is irradiated in the reactor, the nuclide composition changes and, ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is termed burnup credit. Extensive investigations have been performed both within the United States and by other countries in an effort to understand and document the technical issues related to the use of burnup credit.

This Interim Staff Guidance (ISG) provides recommendations to the staff for accepting, on a design-specific basis, a burnup credit approach in the criticality safety analysis of pressurized water reactor (PWR) SNF storage and transportation systems. This revision to ISG-8 incorporates the results of burnup credit-related research that has been conducted since Revision 2 (Rev. 2) was published in September 2002. Based on the detailed results of this research and the technical judgment of the U. S. Nuclear Regulatory Commission (NRC) staff, ISG-8, Rev. 3, includes two major changes in the recommendations to staff reviewing burnup credit applications for SNF transportation and storage systems: (1) optional credit for fission product and minor actinide neutron absorbing isotopes in the SNF composition, and (2) misload analyses and additional administrative procedures in lieu of a burnup measurement at the time of loading. This ISG revision also includes an increase in the maximum assembly average burnup recommended for burnup credit.

Appendix A, *Technical Recommendations for the Criticality Safety Review of PWR Storage and Transportation Casks that Use Burnup Credit*, provides more information on the technical bases for the changes described above. The NRC staff will issue additional guidance and/or recommendations as more information is obtained from research programs directed at burnup credit and as experience is gained through future licensing activities. Except as specified in the Recommendations section of this ISG, the application of burnup credit does not alter the current guidance and recommendations provided by the NRC staff for criticality safety analysis of SNF transportation packages and storage casks.

The guidance provided in this ISG represents one methodology for demonstrating compliance with the criticality safety requirements in 10 CFR Parts 71 and 72 using burnup credit. Following this guidance, the reviewer should be able to determine whether the applicant has provided reasonable assurance that the storage or transportation system meets the applicable criticality safety regulations in 10 CFR Parts 71 and 72. Alternative methodologies proposed by applicants and licensees should be considered on a case-by-case basis, using this guidance to the extent practicable.

Regulatory Basis:

A fissile material transportation package application must demonstrate that a single package is subcritical with water in-leakage. [10 CFR 71.55(b)]

A fissile material transportation package application must demonstrate that a single package is subcritical under normal conditions of transport and hypothetical accident conditions. [10 CFR 71.55(d) and (e), 71.71, and 71.73]

A fissile material transportation package application must demonstrate that arrays of packages are subcritical under normal conditions of transport and hypothetical accident conditions. [10 CFR 71.59, 71.71, and 71.73]

An application for an Independent Spent Fuel Storage Installation must demonstrate that the system meets the criteria for nuclear criticality safety. [10 CFR 72.124]

An application for a spent fuel storage cask must demonstrate that spent fuel is maintained subcritical under credible conditions. [10 CFR 72.236(c)]

Applicability:

This revision to ISG-8 supersedes Revisions 0, 1, and 2 of the ISG in their entirety.

The recommendations that follow were developed with intact fuel as the basis but may also be applicable to fuel that is not intact. If burnup credit is requested by an applicant for fuel that is not intact, the recommendations of this guidance should be applied, as appropriate, to account for uncertainties that can be associated with fuel that is not intact and establish an isotopic inventory and assumed fuel configuration for normal and accident conditions that bounds the uncertainties. Rev. 2 of ISG-1, *Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function*,[3] provides guidance for classifying the condition of the fuel (damaged, undamaged, intact) for SNF storage and transportation.

Recommendations:

1. Limits for the Licensing Basis

   Available data support allowance for burnup credit where the licensing safety analysis is based on major actinide compositions only (i.e., actinide-only burnup credit) or limited actinide and fission product compositions (see Tables A-1 and A-2 of Appendix A) associated with uranium dioxide (UO_2) fuel irradiated in a PWR up to an assembly-average burnup value up to 60 gigawatt-days per metric ton uranium (GWd/MTU) and cooled out-of-reactor for a time period between 1 and 40 years. The range of available measured assay data for irradiated UO_2 fuel supports an extension of the licensing basis up to 5.0 weight percent enrichment in ^{235}U.

   Within this range of parameters, the reviewer should exercise care in assessing whether the analytic methods and assumptions used are appropriate, especially near the limits of the parameter ranges recommended in this ISG for the licensing basis. Use of actinide and fission product compositions associated with burnup values or cooling times outside these specifications should be accompanied by the measurement data and/or justified extrapolation techniques necessary to extend the isotopic validation and quantify or bound the bias and bias uncertainty. Credit for neutron absorbing isotopes other than those identified in Tables A-1 and A-2 of Appendix A should be accompanied by analyses to determine the additional depletion and criticality code bias and bias uncertainty associated with these isotopes.

A certificate or license condition indicating the time limit on the validity of the burnup credit analysis may be necessary in light of the potential need for extended dry storage. Such a condition would depend on the type of burnup credit and the credited post-irradiation decay time.

2. Licensing-Basis Model Assumptions

The actinide and fission product compositions used to determine a value of k-effective (k_{eff}) for the licensing basis should be calculated using fuel design and reactor operating parameter values that appropriately encompass the range of design and operating conditions for the proposed contents. The calculation of the k_{eff} value should be performed using system models and analysis assumptions that allow accurate representation of the physics in the system, as discussed in Section 4 of Appendix A. Attention should be given to the need to:

- account for and effectively model the axial and horizontal variation of the burnup within a SNF assembly (e.g., the selection of the axial burnup profiles, number of axial material zones);
- consider the potential for increased reactivity due to the presence of burnable absorbers or control rods (fully or partially inserted) during irradiation; and
- account for the irradiation environment factors to which the proposed assembly contents were exposed, including fuel temperature, moderator temperature and density, soluble boron concentration, specific power, and operating history.

YAEC-1937, *Axial Burnup Profile Database for Pressurized Water Reactors*,[4] provides a source of representative data that can be used for establishing profiles to use in the licensing basis safety analysis. However, care should be exercised when reviewing profiles intended to bound the range of potential k_{eff} values for the proposed contents for each burnup range, particularly near the upper end of the licensing basis parameter ranges stated in this ISG. NUREG/CR-6801, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*,[5] provides additional guidance on selecting axial profiles.

A licensing basis modeling assumption where the assemblies are exposed during irradiation to the maximum (neutron absorber) loading of burnable poison rods (BPRs) for the maximum burnup encompasses all assemblies that may or may not have been exposed to BPRs. Such an assumption in the licensing basis safety analysis should also encompass the impact of exposure to fully inserted or partially inserted control rods in typical domestic PWR operations. Assemblies exposed to atypical insertions of control rods (e.g., full insertion for one full cycle of reactor operation) should not be loaded unless the safety analysis explicitly considers such operational conditions. If the assumption on BPR exposure is less than the maximum for which burnup credit is requested, then a justification commensurate with the selected value should be provided by the applicant. For example, the lower the exposure, the greater the need to: 1) support the assumption with available data, 2) indicate how administrative controls would prevent a misload of an assembly exposed beyond the assumed value, and 3) address such misloads in a misload analysis.

For assemblies exposed to integral burnable absorbers, the appropriate analysis assumption for absorber exposure varies depending upon burnup and absorber material. The appropriate assumption may be to neglect the absorber while maintaining the other assembly parameters (e.g., enrichment) the same for some absorber materials or for exposures up to moderate burnup levels (typically 20 - 30 GWd/MTU). Thus, a safety analysis including assemblies with integral burnable absorbers should include justification of the absorber exposure assumptions used in the analysis. For assemblies exposed to flux suppressors (e.g., hafnium suppressor inserts) or combinations of integral absorbers and BPRs or control rods, the safety analysis should use assumptions that provide a bounding safety basis, in terms of the effect on system k_{eff}, for those assemblies.

The licensing basis evaluation should include analyses that use irradiation conditions that produce bounding values for k_{eff}, as discussed in Section 4 of Appendix A. The bounding conditions may differ for actinide-only burnup credit versus actinide-plus-fission product burnup credit, and may depend on the population of fuel intended to be loaded in the system (e.g., all PWR assemblies versus a site-specific population). Loading limitations tied to the actual operating conditions may be needed unless the operating condition values used in the licensing basis evaluation can be justified as those that produce the maximum k_{eff} values for the anticipated SNF inventory.

3. Code Validation – Isotopic Depletion

A depletion computer code is used to determine the concentrations of the isotopes important to burnup credit. To ensure accurate criticality calculation results, the selected code should be validated and the bias and bias uncertainty of the code should be determined at a 95% probability, 95% confidence level. Specifically, selection of the code and code validation approach for the fuel depletion analysis should include the considerations in the following paragraphs.

The selected depletion code and cross section library should be capable of accurately modeling the fuel geometry and the neutronic characteristics of the environment in which the fuel was irradiated. Two-dimensional depletion codes have been effectively used in burnup credit analyses. Although one-dimensional codes have been used in some applications, and suffice for making assembly average isotopic predictions for fuel burnup, they are limited in their ability to model increasingly complex fuel assembly designs, and generally produce larger bias and bias uncertainty values because of the approximations necessary in the models. Section 4 of Appendix A provides detailed discussions of the modeling considerations for the code validation analyses.

The destructive RCA data selected for code validation should include detailed information about the SNF samples. This information should include the pin location in the assembly, axial location of the sample in the pin, any exposure to strong absorbers (control rods, BPRs, etc.), the boron letdown, moderator temperature, specific power, and any other cycle-specific data for the cycles in which the sample was irradiated. Note that some RCA data are not suitable for depletion code validation because the depletion histories or environments of these samples are either difficult to accurately define in the code benchmark models, or are unknown. NUREG/CR-7108, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic*

Composition Predictions,[6] provides a recommended set of RCA data suitable for depletion code validation.

The selected code validation approach should be adequate for determining the bias and bias uncertainty of the code for the specific application. The burnup credit analysis results should be adjusted using the bias and bias uncertainty determined for the fuel depletion code, accounting for any trends of significance with respect to different control parameters such as burnup/enrichment ratio or $^{235}U/^{239}Pu$ ratio. NUREG/CR-6811, *Strategies for Application of Isotopic Uncertainties in Burnup Credit*,[7] provides several methodologies that are acceptable to the staff for isotopic depletion validation, including the isotopic correction factor, direct difference, and Monte Carlo uncertainty sampling methods. Section 4 of Appendix A provides detailed discussions of the advantages and disadvantages of these methods. In general, the isotopic correction factor method is considered to be the most conservative, since individual nuclide composition uncertainties are represented as worst-case. The direct difference method provides a realistic "best estimate" of the depletion code bias and bias uncertainty, in terms of Δk_{eff}. The Monte Carlo uncertainty sampling method is more complex and computationally intensive than the other methods, but provides a way to make use of limited measurement data sets for some nuclides. Detailed descriptions of the direct difference and Monte Carlo uncertainty sampling methods are provided in NUREG/CR-7108.

In lieu of an explicit benchmarking analysis, the applicant may use the bias (β_i) and bias uncertainty (Δk_i) values estimated in NUREG/CR-7108 using the Monte Carlo uncertainty sampling method, as shown in Tables 1 and 2 below. These values may be used directly, provided that:

- 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 -VII cross section library),
- the applicant uses appropriate initial assumptions and code input parameters, as discussed in Appendix A,
- the applicant can justify that its design is similar to the hypothetical GBC-32[8] system design used as the basis for the NUREG/CR-7108 isotopic depletion validation, and
- credit is limited to the specific nuclides listed in Tables A-1 and A-2 of Appendix A.

Section 5 of Appendix A provides detailed discussions of the technical basis for the restrictions on directly applying the β_i *and* Δk_i values. β_i values should be added to the calculated system k_{eff}, while Δk_i values may be statistically combined with other independent uncertainties.

Table 1. 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 Range (GWd/MTU)	**Actinides Only Δk_i**	**Actinides and Fission Products Δk_i**
0-5	0.0145	0.0150
5-10	0.0143	0.0148
10-18	0.0150	0.0157
18-25	0.0150	0.0154
25-30	0.0154	0.0161
30-40	0.0170	0.0163
40-45	0.0192	0.0205
45-50	0.0192	0.0219
50-60	0.0260	0.0300

Table 2. 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 Range (GWd/MTU)	**β_i for Actinides and Fission Products**	**Δk_i for Actinides and Fission Products**
0-10	-0.0001	0.0135
10-25	-0.0029	0.0139
25-40	-0.0040	0.0165

4. Code Validation – K_{eff} Determination

Actinide-only credit

Actinide credit should be limited to the specific nuclides listed in Table 1 of Appendix A. Criticality validation for these actinides should be based on the critical experiments available in NUREG/CR-6979, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*,[9] also known as the HTC data, supplemented by mixed-oxide (MOX) critical experiments as appropriate. NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions*,[10] contains a detailed discussion of available sets of criticality validation data for actinide isotopes, and the relative acceptability of these sets. Note that NUREG/CR-7109 demonstrates that fresh UO_2 experiments are not applicable to burned fuel compositions.

Determination of the bias and bias uncertainty associated with actinide-only burnup credit should be performed according to the guidance in NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*.[11] This guidance includes criteria for selection of appropriate benchmark data sets, as well as statistics and trending analysis for determination of criticality code bias and bias uncertainty. An example of bias and bias uncertainty determination for actinide-only burnup credit is included in Section 6 of NUREG/CR-7109.

Fission product and minor actinide credit

The applicant may credit the minor actinide and fission product nuclides listed in Table 2 of Appendix A, provided the bias and bias uncertainty associated with the major actinides is determined as described above. 1.5% of the worth of the minor actinides and fission products conservatively covers the bias due to these isotopes. Due to the conservatism in this value no additional uncertainty in the bias needs to be applied. This estimate is appropriate provided the applicant:

- uses the SCALE code system with the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries,
- uses appropriate initial assumptions and code input parameters, as discussed in Appendix A,
- can justify that its design is similar to the hypothetical GBC-32[8] system design used as the basis for the NUREG/CR-7109 criticality validation, and
- demonstrates that the combined minor actinide and fission product worth is no greater than 0.1 in k_{eff}.

For well qualified, industry standard code systems other than SCALE with the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries, a conservative estimate for the bias associated with minor actinide and fission product nuclides of 3.0% of their worth may be used. Use of a minor actinide and fission product bias less than 3.0% should be accompanied by additional justification that the lower value is an appropriate estimate of the bias associated with that code system.

5. Loading Curve and Burnup Verification

Burnup credit evaluations should include loading curves which specify the minimum required assembly average burnup as a function of initial enrichment for the purpose of loading SNF storage or transportation systems. Separate loading curves should be established for each set of applicable licensing conditions. For example, a separate loading curve should be provided for each minimum cooling time to be considered in the system loading. The applicability of the loading curve to bound various fuel types or burnable absorber loadings should be justified.

Burnup verification should be performed to ensure that a storage or transportation system evaluated using burnup credit is not loaded with an assembly more reactive than those included in the loading criteria. Verification should include a measurement that confirms the reactor record for each assembly. Confirmation of reactor records using measurement of a sample of fuel assemblies will be considered if the sampling method can be justified in comparison to measuring every assembly.

The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement. NUREG/CR-6998, *Review of Information for Spent Nuclear Fuel Burnup Confirmation*,[12] contains bounding estimates of reactor record burnup uncertainty.

Measurements should be correlated to reactor record burnup, enrichment, and cooling time values. Measurement techniques should:

- account for any measurement uncertainty (typical within a 95% confidence interval) in confirming reactor burnup records, and
- include a database of measured data (if measuring a sampling of fuel assemblies) to justify the adequacy of the procedure in comparison to procedures that measure each assembly.

Misload Analyses

Misload analyses may be performed in lieu of a burnup measurement. A misload analysis should address potential events involving the placement of assemblies into a SNF storage or transportation system that do not meet the proposed loading criteria. The applicant should demonstrate that the system remains subcritical for misload conditions, including calculation biases, uncertainties and an appropriate administrative margin that is not less than 0.02 Δk. An adequate justification, that includes 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, should accompany the use of any administrative margin that is less than the normal 0.05 Δk.

A misload analysis should consider:

- misloading of a single severely underburned assembly and,
- misloading of multiple moderately underburned assemblies.

The severely underburned assembly for the single misload analysis should be chosen such that the assembly average burnup and initial enrichment along an equal reactivity curve bound 95% of the discharged fuel population considered unacceptable for loading in a particular storage or transportation system with 95% confidence. The multiple moderately underburned assemblies for this analysis should be assumed to make up at least 50% of the system payload, and should be chosen such that the burnups and initial assembly average enrichments along an equal reactivity curve bound 90% of the total discharged fuel population. The 2002 Energy Information Administration RW-859 fuel survey[13], or a later estimate, is acceptable as an estimate of discharged fuel population characteristics.

The misload analysis should also consider the effects of placing the underburned assemblies in the most reactive positions within the loaded system (e.g., middle of the fuel basket). If removable non-fuel absorbers were credited as part of a criticality safety analysis (e.g., poison rods added to guide tubes), the misload analysis should consider misloading of these absorbers. Additionally, the misload analysis should consider assemblies with greater burnable absorber or control rod exposure than assumed in the criticality analysis, if assumed exposure is not bounding. NUREG/CR-6955, *Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*,[14] illustrates the magnitude of k_{eff} changes that can be expected as a result of various misloads in a theoretical GBC-32 SNF storage and transportation system.

Administrative Procedures

A misload analysis should be coupled with additional administrative procedures to ensure that the SNF storage or transportation system will be loaded with fuel that is within the specifications of the approved contents. Procedures considered to protect against misloads in storage and transportation systems that rely on burnup credit for criticality safety may include:

- verification of the location of high reactivity fuel (i.e., fresh or severely underburned fuel) in the spent fuel pool both prior to and after loading,
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement),
- quality assurance audit of the canister or cask inventory and loading records prior to 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 generation of the fuel move instructions, and
- minimum soluble boron concentration in pool water, to offset a potential misload, during loading and unloading.

Recommendation

The staff recommends that the appropriate chapters of NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems*,[15] NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*,[16] and NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*,[17] be revised to address the guidance contained in this ISG and the information contained in the appendix.

Approved by:____ Date: _______________

Mark Lombard, Director, SFST

References:

1. Code of Federal Regulations, Title 10, Part 71, *Packaging and Transportation of Radioactive Material*, January 1, 2012.
2. Code of Federal Regulations, Title 10, Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*, January 1, 2012.
3. U.S. Nuclear Regulatory Commission, Spent Fuel Project Office Interim Staff Guidance – 1, Rev. 2 – *Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function*, U.S. Nuclear Regulatory Commission, May 11, 2007.
4. R. J. Cacciapouti and S. Van Volkinburg, *Axial Burnup Profile Database for Pressurized Water Reactors*, YAEC-1937, May 1997. Available as Data Package DLC-201 from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, http://www-rsicc.ornl.gov/ORDER.html.
5. J. C. Wagner, M. D. DeHart, and C. V. Parks, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*, NUREG/CR-6801 (ORNL/TM-2001/273), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2003.
6. G. Radulescu, I. C. Gauld, G. Ilas, and J. C. Wagner, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions*, NUREG/CR-7108 (ORNL/TM-2011/509),U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, April 2012.

7. I. C. Gauld, *Strategies for Application of Isotopic Uncertainties in Burnup Credit*, NUREG/CR-6811 (ORNL/TM-2001/257), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, June 2003.
8. J. C. Wagner, *Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*, NUREG/CR-6747 (ORNL/TM-2000/306), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, October 2001.
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11. J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, C. M. Hopper, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*, NUREG/CR-6361 (ORNL/TM-13211), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 1997.
12. B. B. Bevard, J. C. Wagner, C. V. Parks, M. Aissa, *Review of Information for Spent Nuclear Fuel Burnup Confirmation*, NUREG/CR-6998 (ORNL/TM-2007/229), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, December 2009.
13. *RW-859 Nuclear Fuel Data*, Energy Information Administration, Washington, D.C., October 2004.
14. J. C. Wagner, *Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*, NUREG/CR-6955 (ORNL/TM-2004/52), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, January 2008.
15. NUREG-1536, *Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*, U.S. Nuclear Regulatory Commission, July 2010.
16. NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*, U.S. Nuclear Regulatory Commission, March 2000.
17. NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, U.S. Nuclear Regulatory Commission, March 2000.

Appendix A: Technical Recommendations for the Criticality Safety Review of PWR Transportation Packages and Storage Casks That Use Burnup Credit

1. Introduction

The overall reactivity decrease of nuclear fuel irradiated in light water reactors is due to the combined effect of the net reduction of fissile nuclides and the production of parasitic neutron absorbing nuclides (non-fissile actinides and fission products). Burnup credit refers to accounting for partial or full reduction of spent nuclear fuel (SNF) reactivity caused by irradiation. This Interim Staff Guidance (ISG) provides guidance to the staff for its use in the review of spent fuel cask designs that seek burnup credit. This Appendix provides the technical bases for the recommendations provided in the ISG.

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 U.S. Nuclear Regulatory Commission (NRC) Spent Fuel Project Office (SFPO) issued ISG-8, Revision 2[1] 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, Rev. 2, was published, SFPO staff recommended actinide-only credit. Additionally, 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 which credit burnup in the criticality analysis.

Since ISG-8, Rev. 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 report will summarize 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 staff read the referenced reports and papers to understand the detailed evaluation of specific burnup credit parameters discussed in this report. A comprehensive bibliography of burnup credit-related technical reports and papers is provided at http://www.ornl.gov/sci/nsed/rnsd/pubs_burnup.shtml.

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 required 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:

1) additional information and assumptions for input to the analysis,
2) additional analyses to obtain the SNF compositions,
3) additional validation efforts for the depletion and decay software,

4) enhanced validation to address the additional nuclides in the criticality analyses, and
5) verification that the fuel assembly to be loaded meets the minimum burnup requirements made prior to 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 licenses and Certificates of Compliance (CoCs) have generally shifted to high capacity designs (i.e., 32 fuel assemblies or greater) in the past decade. 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 which 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 this ISG include:

1) general information on limits for the licensing basis,
2) recommended assumptions regarding reactor operating conditions,
3) guidance on code validation with respect to the isotopic depletion evaluation,
4) guidance on code validation with respect to the k_{eff} evaluation, and
5) guidance on preparation of loading curves, and the process for assigning a burnup loading value to an assembly.

Each of these five areas should be considered in a criticality safety analysis that uses burnup credit.

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 ISG and justify any exceptions taken due to the nature of the fuel (e.g., the use of an axial profile that is not consistent with the recommendation). Rev. 2 of ISG-1, *Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function*,[2] provides guidance for classifying the condition of the fuel (e.g., damaged, intact) for SNF storage and transportation.

The validation methodologies presented in Sections 4 and 5 of this document were performed for a representative cask model, known as the GBC-32, described in NUREG/CR-6747, *Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*.[3] As will be discussed later in this report, in order to directly use bias and bias uncertainty numbers developed in:

1) NUREG/CR-7108, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions*,[4] and
2) NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions*,[5]

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, reflector materials and dimensions, etc. 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), 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 of this ISG 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 due to 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. Although the ISG does not provide explicit guidance on BWR burnup credit, criticality analyses which include such credit should be reviewed on a case-by-case basis.

The remainder of this report 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).

3. Limits for Licensing Basis (Recommendation 1)

Available validation data supports actinide-only and actinide and fission product burnup credit for UO_2 fuel enriched up to 5.0 weight percent ^{235}U, 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.

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*.[6] This report concludes that the actinides and fission products listed in Tables A-1 and A-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 will be discussed in Sections 5 and 6 of this Appendix. Accurate prediction of the concentrations for the nuclides in Tables A-1 and A-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 A-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 A-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 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 sufficient penalties to account for the lack of such data.

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 ^{235}U initial enrichment. Though limited RCA data is 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 ^{235}U.

Cooling Time

Figure A-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 time frame 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.[7] In light of the increasingly likely scenario of extended dry storage of SNF, the CoC for a SNF transportation package 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 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, *Recommendation on the Credit for Cooling Time in PWR Burnup Credit Analysis*,[8] provides a comprehensive study of the effect of cooling time on burnup credit for various cask designs and SNF compositions.

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 ISG 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.

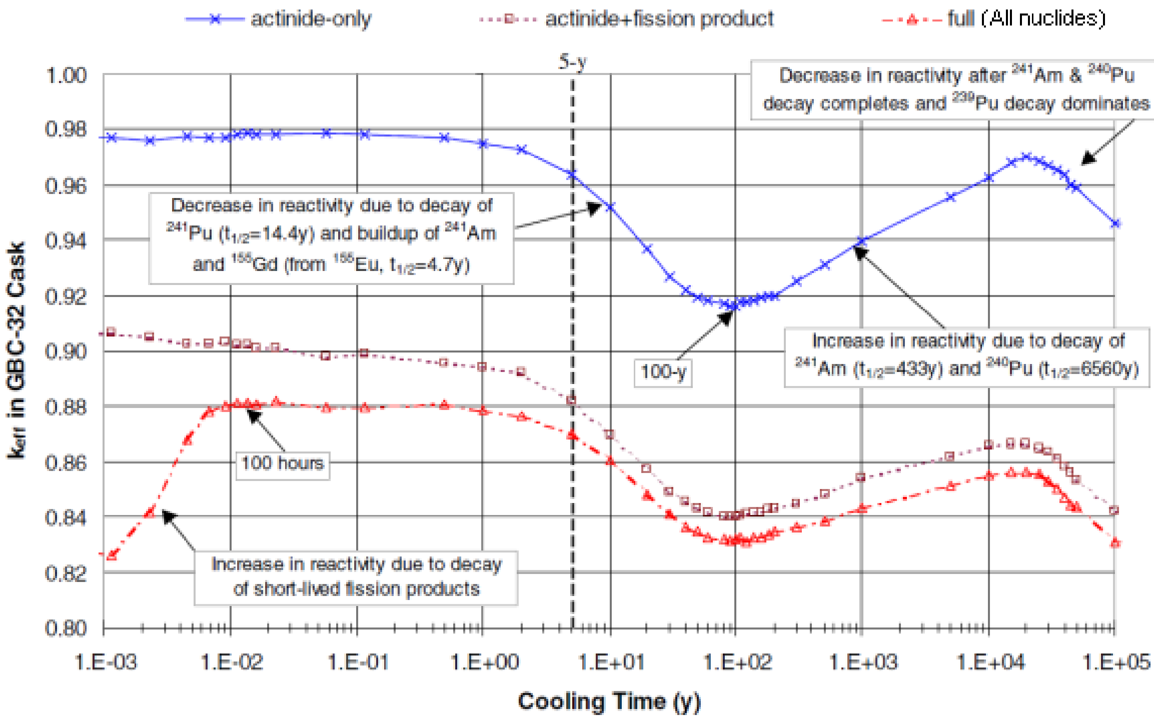

Figure A-1: Reactivity behavior in the GBC-32 cask as a function of cooling time for fuel with 4.0 wt % ^{235}U initial enrichment and 40 GWd/MTU burnup[7]

4. **Licensing-Basis Model Assumptions (Recommendation 2)**

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., W17x17 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.

Reactor Operating History and Parameter Values

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 are described in Section 4.2 of NUREG/CR-6665.

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 A-2 to A-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 due to the parameter increase causing increased production of fissile plutonium nuclides and decreased ^{235}U utilization.

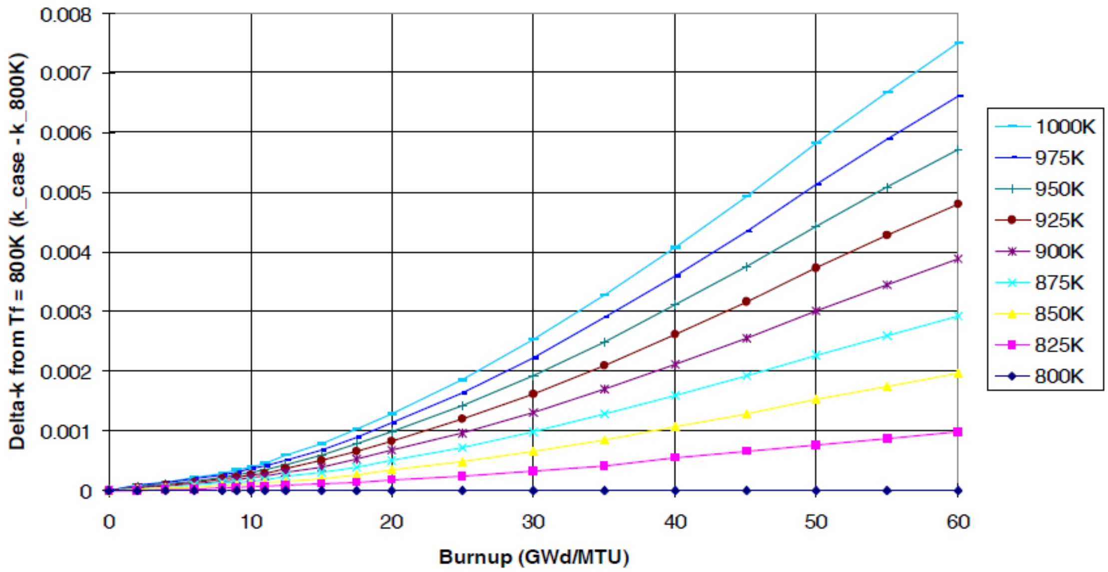

Figure A-2: Reactivity effect of fuel temperature during depletion on k_{inf} in an array of poisoned storage cells. Results correspond to fuel with 5.0 weight percent initial ^{235}U enrichment.[9]

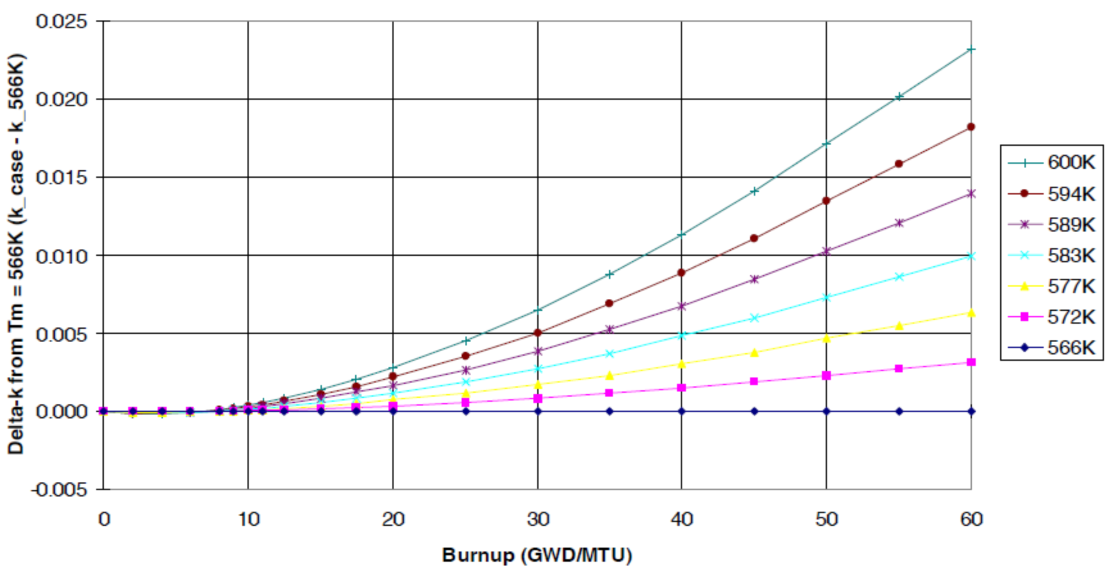

Figure A-3: Reactivity effect of moderator temperature during depletion on k_{inf} in an array of poisoned storage cells. Results correspond to fuel with 5.0 weight percent initial ^{235}U enrichment.[9]

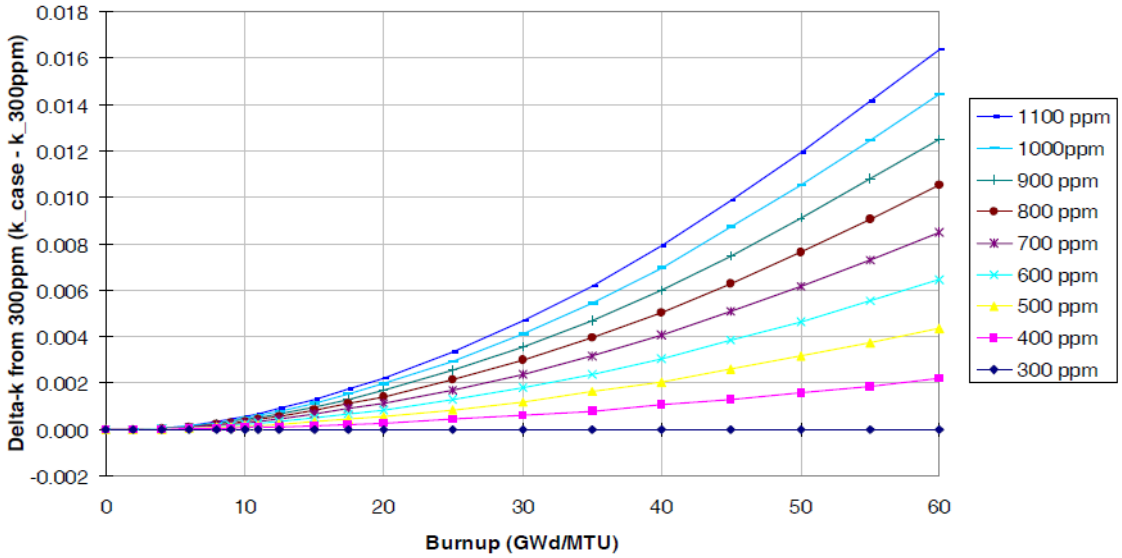

Figure A-4: Reactivity effect of soluble boron concentration during depletion on k_{inf} in an array of poisoned storage cells. Results correspond to fuel with 5.0 weight percent initial ^{235}U enrichment.[9]

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 A-5 and A-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 Ref. 10). 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.

More detailed information on the impact of each parameter or phenomenon that should be assumed in the depletion model is provided in Refs. 6 and 10. 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*[11]).

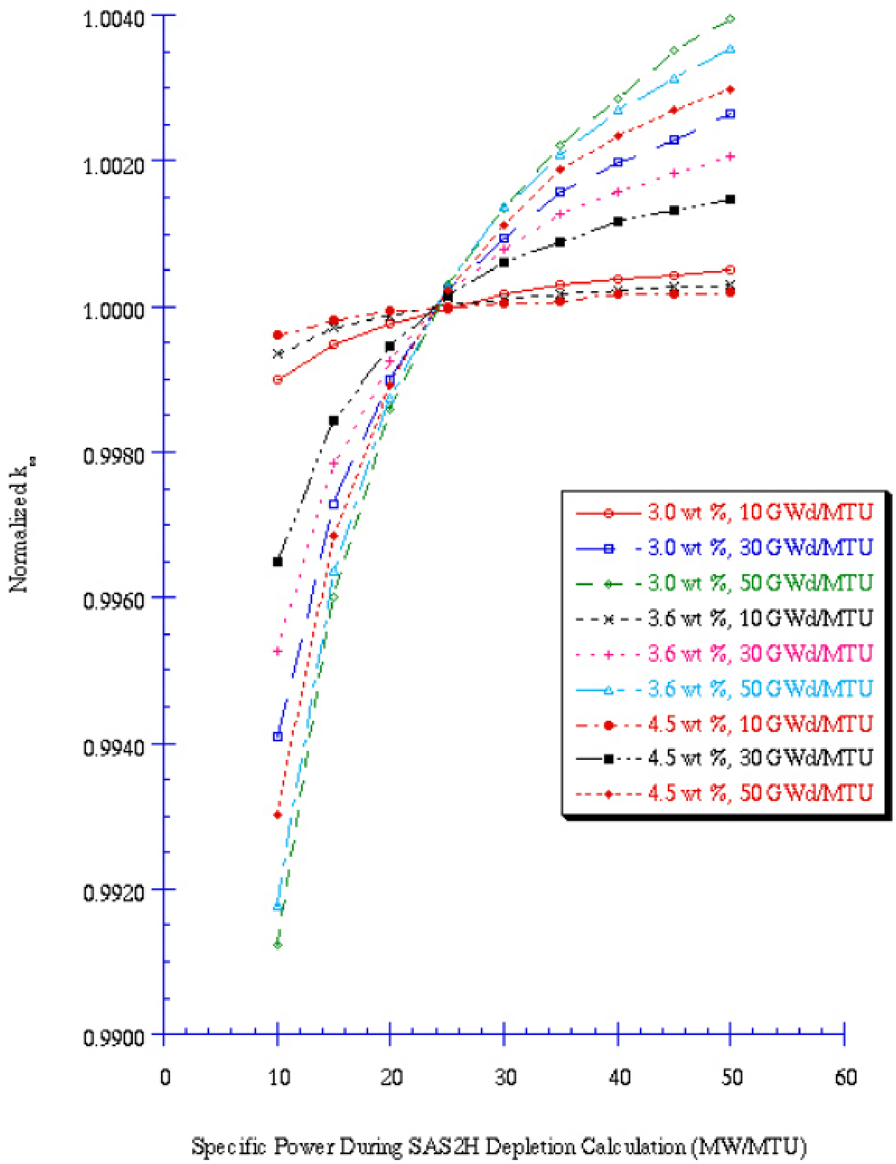

Figure A-5: Reactivity effect of specific power during depletion on k_{inf} in an array of fuel pins (actinides only).[10]

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.

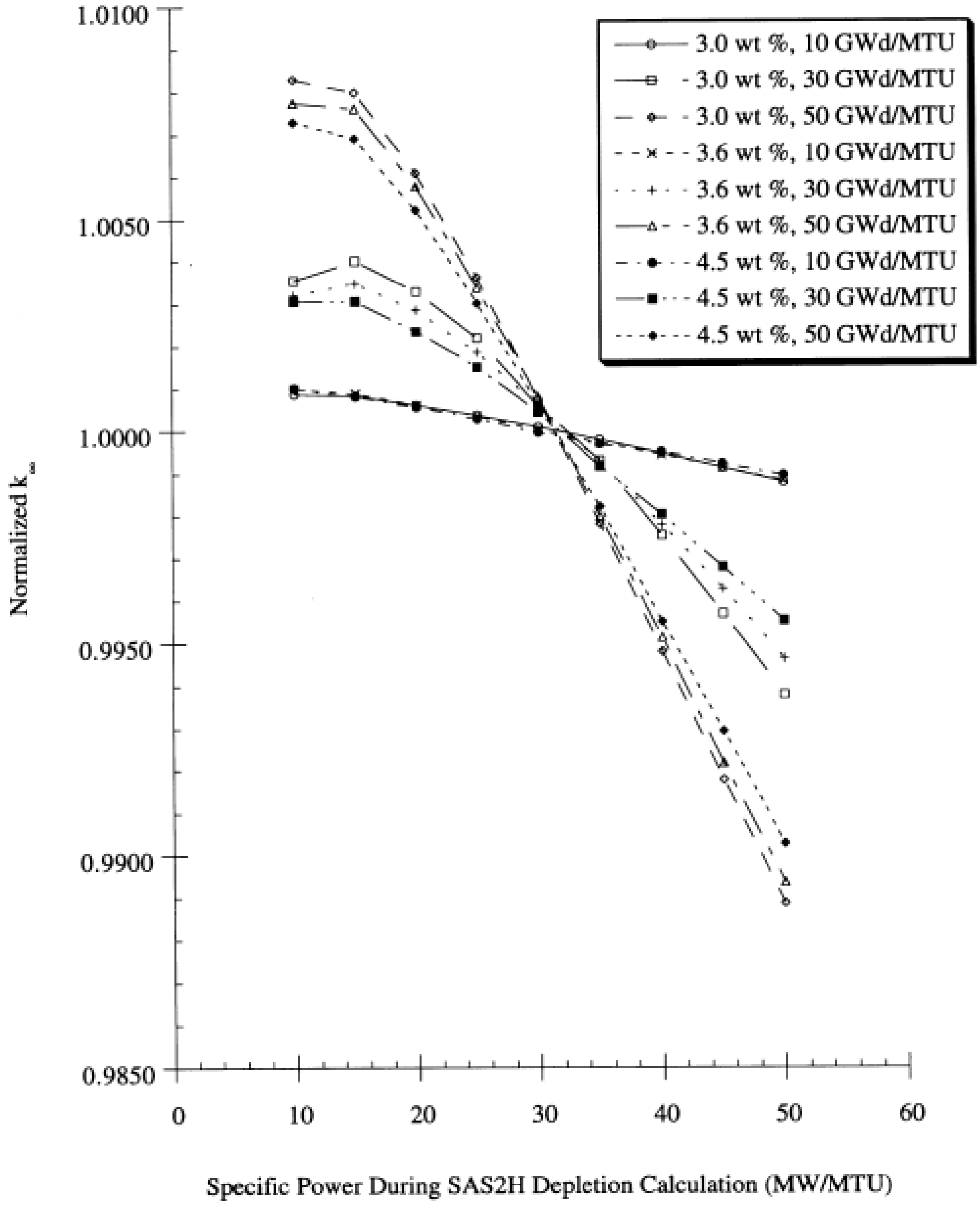

Figure A-6: Reactivity effect of specific power during depletion on k_{inf} in an array of fuel pins (actinides and fission products).[10]

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.[12] In large rail casks, the probability that underburned quadrants of multiple fuel assemblies will be oriented in such a way as to have a substantial impact of k_{eff} is not expected to be significant. The safety evaluation should address the impact of horizontal burnup gradients such as found in Ref. 12 on their cask design or demonstrate that the assemblies to be loaded in the cask will be verified to not have such gradients. One acceptable approach would be to determine the difference in k_{eff} for a cask loaded with fuel having a horizontal burnup gradient and a cask loaded with the same fuel having a uniform horizontal burnup (i.e., no gradient). The fuel with the gradient would be arranged so as to maximize the reactivity effect of the gradient. The reactivity difference between the two cases could then be applied to the remaining analyses.

Axial Burnup Profiles

Considerable attention should be paid to the axial burnup profile(s) selected for use in the safety evaluation. A uniform axial profile is generally bounding at low burnups but is increasingly non-conservative at higher burnups due to the increasing relative worth of the fuel ends, as demonstrated in NUREG/CR-6801, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*.[13] Figure A-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 non-conservative at higher burnups. The burnup range at which this transition occurs will vary with fuel design and the type of burnup credit.

This ISG indicates 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 staff should consider the range of profiles anticipated for the fuel to be loaded in the system.

The publicly available database of axial profiles in YAEC 1937, *Axial Burnup Profile Database for Pressurized Water Reactors*,[14] 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% 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 ^{235}U, 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 ^{235}U, if profiles are selected that include a margin for the potential added uncertainty in moving to the higher burnups and initial enrichments allowed per the ISG. 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.

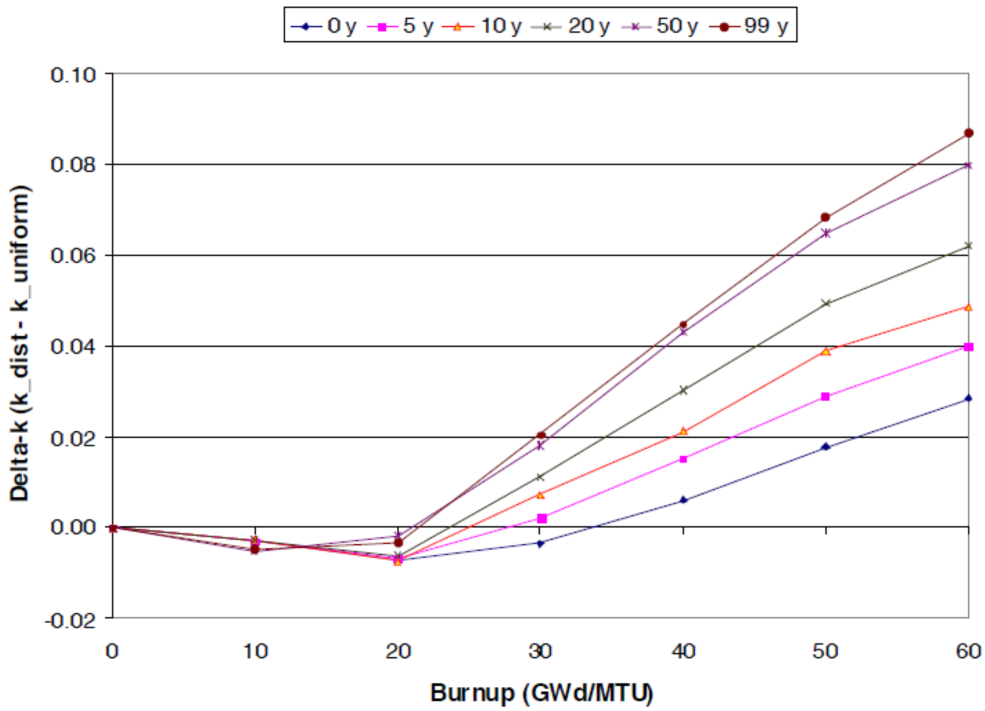

Figure A-7: Effect of axial burnup distribution on k_{eff} in the GBC-32 cask for actinide-only burnup credit and various cooling times for fuel with 4.0 weight percent initial enrichment.[9]

While the preceding discussion indicates that the database is an appropriate source of axial burnup profiles, the staff 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 the database similar to that demonstrated 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 it is 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 prior to 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.

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 which are not exposed. This is due to the hardening of the neutron spectrum, and will lead to increased fissile plutonium nuclide production and reduced ^{235}U depletion. In addition, when removable neutron absorbers are inserted, the spectrum is further hardened due to displacement of the moderator. NUREG/CR-6761, *Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit*,[15] and NUREG/CR-6760, *Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit*,[16] provide characterizations of the effects of burnable absorbers on spent fuel. 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 and/or for less than the maximum burnup (for which credit is requested) 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 upon the material type and the burnup level. Exposure to the maximum absorber loading was seen to be bounding for zirconium diboride (ZrB_2)-type IBAs (known as integral fuel burnable absorbers, or IFBAs) 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 ISG, whether fixed, removable, or a combination of the two, should be evaluated on a case-by-case basis.

Control Rods

As with BPRs, CRs fully or partially inserted during reactor operation can harden the spectrum 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 the parametric study of the effects of CR exposure conducted in NUREG/CR-6759, *Parametric Study of the Effect of Control Rods for PWR Burnup Credit*,[17] 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[11] includes a representative sampling of assemblies exposed to CRs and APSRs, the appropriate selection of a limiting axial profile(s) from that database would be expected to adequately encompass the potential impact for axial profile distortion caused by CRs and APSRs.

Exposures to CR or APSR insertions or materials not considered in the references supporting this ISG 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.

Depletion Analysis Computational Model

For depletion analyses, computer codes that can track a large number of nuclides should 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 A-1 and A-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, due to their current limitations.

Several two-dimensional neutron transport theory based codes are available, such as CASMO, HELIOS, and the SCALE TRITON sequence.[18] Staff 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, due to the limited availability of accurate two-dimensional computer codes, most burnup credit calculations used one-dimensional depletion codes to determine spent fuel 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[19]), one-dimensional physics models of PWR assembly designs can produce sufficiently accurate assembly average spent fuel 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 will be discussed in greater detail in Section 5 of this Appendix.

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 or transportation package are documented in the following staff review guidance documents:

- NUREG/CR-5661, *Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages*,[20]
- NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*,[21]
- NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*,[22]
- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*,[23] and
- NUREG-1536, *Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*.[24]

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% or more) for potential water inleakage. The analyses in Appendix A of NUREG/CR-6801 demonstrate that both modeling the end regions as either 100% steel or full-density water provides a higher value of k_{eff} than a combination (homogenized mixture 50% water and 50% steel assumed) of the two. For the cask that was studied, the all steel reflector provided a k_{eff} change of nearly 1% over that of full density water. Although use of 100% 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 non-uniformly, due to 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 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 a non-trivial process that has the potential for error. The staff should verify the interface process and/or the computer code used to automate the data handling. As with fresh fuel criticality analyses, the staff should verify that the criticality analyses for burnup credit is 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 will be discussed in greater detail in Section 6 of this Appendix.

5. Code Validation – Isotopic Depletion (Recommendation 3)

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

Selection of Validation Data

Validation data consist of measurements of isotopic concentrations from destructive RCA samples of SNF. Reliable depletion code validation results require a sufficient number of data sets that include all isotopes for which burnup credit is taken. The applicant, therefore, should provide justification of the sample size for each nuclide. For example, the applicant should demonstrate that isotopic uncertainty is appropriately increased to account for uncertainty associated with 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 NRC and ORNL in the following references:

- NUREG/CR-7012, *Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel*,[25]
- NUREG/CR-7013, *Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor*,[26]
- NUREG/CR-6968, *Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors*,[27]
- NUREG/CR-6969, *Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-ARIANE and REBUS Programs (UO_2 Fuel)*.[28]

NUREG/CR-7108 analyzed the available data sets and identified 100 fuel samples suitable for depletion code validation for SNF storage and transportation systems. The staff 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.

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, 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 calculation of individual RCA sample nuclide concentrations.

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,[29] 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 a representative SNF storage and transportation system. 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 A-1 and A-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 ^{95}Mo, ^{101}Ru, ^{103}Rh, and ^{133}Cs in NUREG/CR-7108 (see Section 6.2).

Based on the recent 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[30] 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., $A \rightarrow B \rightarrow C$), the content of nuclide B at a reference cooling time can be obtained by solving the Bateman Equation. The time-adjusted isotopic concentration should be used in the validation, rather than the measurement data. In the case where only a fraction of the decay leads to the production of nuclide B, the fraction of decay of nuclide A leading to nuclide B should also be included. For a nuclide without a significant precursor, the contribution from decay of precursors should be set to zero, and only the decay of nuclide B need be accounted for.

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 β_i *and* Δk_i 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 β_i and Δk_i values for the hypothetical GBC-32 storage and transportation system. It is acceptable for the applicant to use the β_i and Δk_i values from Tables 1 and 2 of the ISG 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 -VII cross section library),
- the applicant uses appropriate initial assumptions and code input parameters,
- 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, and
- credit is limited to the specific nuclides listed in Tables A-1 and A-2.

β_i values should be added to the calculated system k_{eff}, while Δk_i 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 and EALF. In case the actual design is significantly different from the GBC-32 cask, or the applicant uses a different code and/or cross section library for its analysis, the applicant should use the direct difference or isotopic correction factor methods discussed previously.

6. Code Validation – K_{eff} Determination (Recommendation 4)

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 ANSI/ANS 8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*,[31] "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, Rev. 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 (MOX) systems. These systems are not entirely representative of SNF in a transportation package, as fresh UO_2 systems contain no plutonium, and the MOX experiments generally do not have plutonium isotopic ratios consistent with that of burned fuel.

While there were no representative critical experiments for SNF transportation 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, staff deemed that it was appropriate to recommend "actinide-only" credit for SNF transportation criticality safety evaluations. This approach represented the bulk of the reduction in k_{eff} due to depletion of the fuel (see Table A-3), and excluded the fission products which served as additional margin to cover uncertainties due to modeling actinide depletion k_{eff} effects.

Although there continue to be insufficient critical experiments for a traditional validation of the code-predicted reduction in k_{eff} due to fission products and minor actinides in spent fuel, a group of critical experiments designed for validating SNF k_{eff} reduction due to major actinides has become available since ISG-8, Rev. 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*,[32] 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} due to 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 to perform a traditional validation for the actinide component of the reduction in k_{eff} due to 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} due to burnup is relatively small (see Table A-3), applicants for SNF transportation packages have requested the additional credit represented by these absorbers. The apparent need for fission product credit is due to the significant increase in percentage of discharged PWR fuel assemblies capable of being shipped in a high capacity (e.g., 32 assembly) rail transportation package. Figure A-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 overlaid on this figure, showing the relative amounts of the PWR fuel population which 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.

The ability to properly validate criticality codes for actinide burnup credit is a crucial step towards recommending fission product credit, as the actinides represent the bulk of the reduction in k_{eff} due to 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 which 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 ^{149}Sm (LEU-COMP-THERM-050), another involving ^{103}Rh (LEU-COMP-THERM-079), and a third involving elemental Sm, Cs, Rh, and Eu (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 spent fuel 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), staff does not expect to see such experiments carried out within a reasonable timeframe. In the absence of such important criticality validation data, staff and their contractors at ORNL sought alternative methodologies for estimating fission product bias and bias uncertainty.

Table A-3: FP Reactivity Worth for "Typical" Burnup in Generic Burnup Credit Cask (GBC-32) with 4 weight percent ^{235}U Westinghouse 17 × 17 OFA, Burned to 40 GWd/MTU

Credited Nuclides	k_{eff}	Δk	$\%\Delta k$[1]
Fresh Fuel	1.13653		
8 Major Actinides[2]	0.94507	0.19146	71.9
All Actinides	0.93486	0.01021	3.8
Key 6 Fission Products[3]	0.88499	0.04987	18.7
All Remaining Fission Products	0.87010	0.01489	5.6
Total		0.26643	100

[1]This is the percent of total Δk for the burnup attributable to the portion of the total nuclide population in the first column
[2]8 major actinides include ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am
[3]Key 6 fission products include ^{103}Rh, ^{133}Cs, ^{149}Sm, ^{151}Sm, ^{143}Nd, and ^{155}Gd

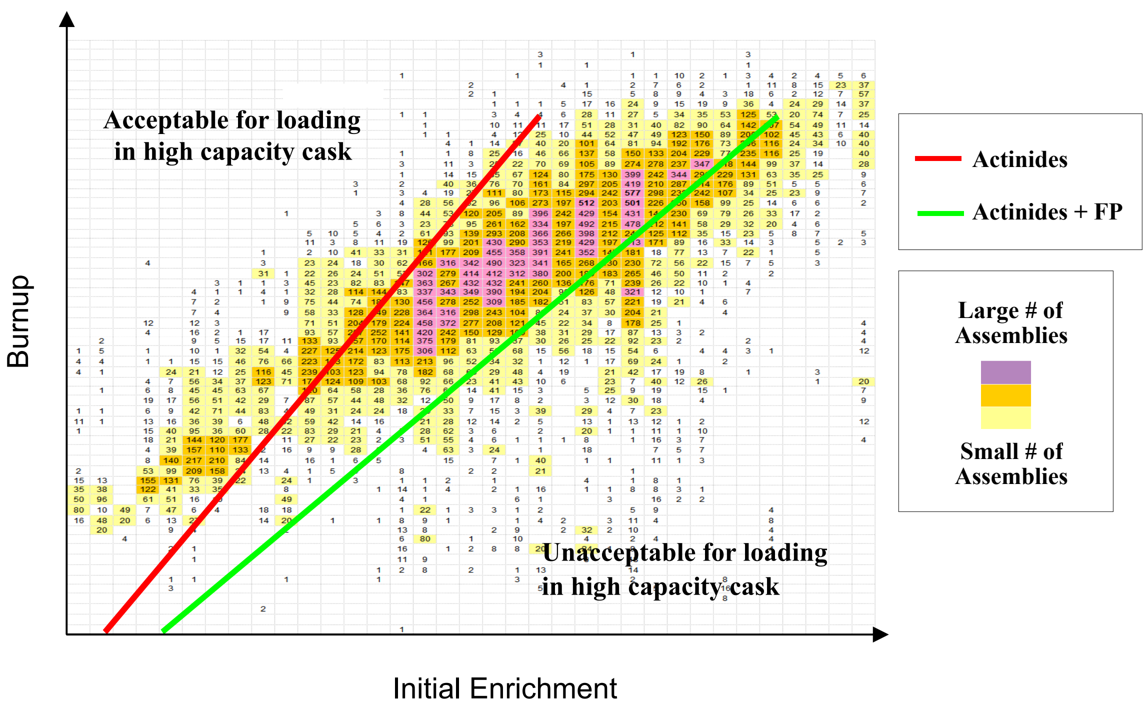

Figure A-8: Representative Loading Curves and Discharged PWR Population

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,[33] 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. The validity of this theoretical basis is discussed in greater detail in NUREG/CR-7109.

This methodology has been benchmarked against a large number of low enrichment uranium (LEU) critical experiments, high enrichment uranium (HEU) 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. The uncertainty analysis method is described and details of the comparisons are provided in NUREG/CR-7109. The results demonstrate that, for a generic SNF transportation package evaluated with the SCALE code system, and the ENDF/B-V, -VI, or -VII cross section libraries, the total fission product nuclear data uncertainty (1σ) does not exceed 1.5% of the total minor actinide and fission product worth for the 19 nuclides (Table A-2) considered over the burnup range of interest (i.e., 5 to 60 GWd/MTU). Since the uncertainty in k_{eff} due to uncertainty in the cross section data is an indication of how large the actual code bias could be, the 1.5% value should be used as a bias (i.e., added directly to the calculated k_{eff}). Due to the conservatism in this value no additional uncertainty in the bias needs to be applied.

In order to use the 1.5% 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, -VI or -VII cross section data. In this case, the combined minor actinide and fission product bias and bias uncertainty should be increased to 3.0%. NUREG/CR-7109 shows that the bias and bias uncertainty is 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% determined for the SCALE code system is conservative. Staff should consider applicant requests to use the 1.5% 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).

Integral Validation

ANSI/ANS 8.27-2008, *Burnup Credit for LWR Fuel*,[34] 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 source of integral validation data are commercial reactor critical (CRC) state points. These CRC state points consist of either a hot zero-power critical condition attained after sufficient cooling time to allow the fission product xenon inventory to decay or at-power equilibrium critical condition where xenon worth has reached a fairly stable value.

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

7. Loading Curve and Burnup Verification (Recommendation 5)

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 A-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 due to misidentification, mischaracterization, or misplacement of fuel assemblies. This has resulted in unanalyzed loading configurations during storage of spent fuel 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, due to fabrication or performance issues, some fuel assemblies have been removed from the reactor before achieving their desired burnup. 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, Rev. 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 ISG-8, 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, etc.), and visually identifiable. Finally, there is an economic incentive involved with new fuel assemblies which 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 a 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. A majority of these misloads occurred as a result of operator errors or inaccurate parameter data (i.e., burnup, enrichment, cooling time). Using available misload data, the NRC Office of Nuclear Regulatory Research (RES), in a report titled *Estimating the Probability of Misload in a Spent Fuel Cask*,[36] evaluated the likelihood of misloading fuel assemblies within a spent fuel 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*,[37] 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 ^{235}U, while placing between one and four misloaded assemblies into the most reactive positions within the system. All assemblies within the system, with the exception of the misloaded assemblies, were assumed to undergo a cooling period of 5 years. The misloaded assemblies were evaluated at 90, 80, 50, 25, 10, and 0% (unirradiated) of the minimum assembly average burnup value required by the loading curve.

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

Misload Evaluation

The applicant's misload evaluation should be based on a reliable and relatively recent estimate of the discharged PWR fuel population, and should reflect the segment of that population that is intended to be stored or transported in the 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., W17x17 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 Survey,[38] 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% of the discharged fuel population considered unacceptable for loading in a particular storage or transportation system with 95% 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 the next section), 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. % U-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% of the system payload, and should be chosen such that the assembly average burnups and initial enrichments along the equal reactivity curve bound 90% 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% 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 which bounds 90% 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% 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% 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% 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% 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% of the cask capacity would bound the extent of likely multiple-misload conditions. The implementation of the administrative procedures recommended in this ISG 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% 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 ensure against unknown errors or uncertainties in the method of calculating k_{eff} as well as impacts of system design and operating conditions not explicitly considered in the analysis. 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[39] 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.

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:

- verification of the location of high reactivity fuel (i.e., fresh or severely underburned fuel) in the spent fuel pool both prior to and after loading,
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement),
- quality assurance audit of the system inventory and loading records prior to 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, and
- minimum required soluble boron concentration in pool water during loading and unloading.

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 quality assurance audit of the system inventory and loading records is intended to verify that the contents of previously loaded systems are as expected prior to shipment. 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 a single, fresh 5.0 weight percent enriched fuel assembly in the worst location in the system.

Misload analyses are included in this revision of ISG-8 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 spent fuel transportation cask. Using credible bounding assumptions, a misload analysis could be generated to account for potential events involving loading, while maintaining an appropriate safety margin.

8. References

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APPENDIX B PUBLIC COMMENTS RECEIVED AND THEIR DISPOSITION

The purpose of this appendix is to list all the public comments received on Interim Staff Guidance 8 (ISG-8), “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks,” Revision 3. The NRC issued ISG-8, Revision 3 (ML12115A303) for public comment on May 2, 2012 for a 30 day period and received comments from the following three sources:

- NEI, Nuclear Energy Institute, letter from Marcus Nichol to Ms. Cindy Bladey, USNRC, dated May 30, 2012 (ML12156A265)
- NuclearConsultants.com, letter from Mr. Dale Lancaster to USNRC, dated May 30, 2012 (ML12156A266)
- Holtec International, letter from Stefan Anton to USNRC, dated June 1, 2012 (ML12158A189)

The staff’s resolution and any associated changes to the ISG are listed for each comment in the following table.

Comment	Summary of Comment	Resolution	Changes to ISG
NEI 1	Recommend a more risk informed approach to burn-up verification, based primarily upon the use administrative procedures, with the misload analysis providing defense-in-depth, and eliminates burnup measurements. This approach is consistent with current industry practice in accordance with NRC approved cask licensing bases, and is supported by the technical references cited in the draft guidance. We further note that this is the most effective method of addressing the situations that could result in a misload.	The burnup measurement recommendation is maintained as an option in the draft ISG, and does not need to be performed if the applicant performs a misload analysis accompanied by additional administrative procedures. The combination of misload analysis and additional administrative procedures is intended to provide defense-in-depth against cask misloads, and staff does not see the ordering of these two elements as important (i.e., both the misload analysis and the administrative procedures are equally necessary).	None.
NEI 2	Further clarify that the purpose of burn-up verification is to prevent the three credible situations that could result in more reactive assemblies being loaded: 1) loading the wrong fuel assembly, 2) calculating a burn-up value higher than actual, and 3) assigning the wrong burn-up value to a fuel assembly. Expanding upon the concept that the purpose of burn-up verification is to "ensure that a storage or transportation system evaluated using burn-up credit is not loaded with an assembly more reactive than those included in the loading criteria" will provide a clear basis for the guidance related to appropriate methods of burn-up verification, and ensure efficient use of industry and NRC resources in the licensing, implementation, and oversight of these activities.	Staff believes that the way misloads are defined in the draft ISG is adequate and appropriate for consideration in the burnup credit criticality analysis. Further definition is provided in the references to Section 5 of the draft ISG and Section 7 of Appendix A, and is unnecessary in the ISG itself.	None.
NEI 3	Recommend that the administrative	Staff believes that all of the administrative	Revised ISG to clarify that

	procedures in draft revision 3 of ISG-8 be replaced by the following: #1. Verify the identity of the fuel assembly prior to loading it into the cask #2. Verify the identity of the fuel assemblies loaded into the cask prior to closing #3. Verify the burn-up values of each fuel assembly to be loaded into the cask from a source QA record prior to loading the first assembly #4. Reduce the verified reactor record burn-up value by uncertainty in the record value, this is the burn-up value to be used for loading acceptance #5. Verify that each fuel assembly to be loaded into the cask satisfies the loading requirements prior to loading the first assembly #6. Develop and perform procedures/ processes in accordance with the QA program #7. Verify that the soluble boron concentration in the pool and cask is greater than the minimum required prior to cask loading	procedures recommended by NEI should already be reflected in either the site or cask operating procedures, even for systems that do not credit burnup in the criticality safety analysis. The list of procedures in the draft ISG are recommended *additional* administrative procedures for the loading of a burnup credit cask, which staff believes will reduce the likelihood or consequences of a high-reactivity misload. Note that the RES report on misload probability demonstrates that misloads are credible with the procedures included in the above list. Staff will revise the ISG to clarify that the recommended administrative procedures are *in addition* to those that would normally be in place for a non-burnup credit cask system.	procedures are in addition to those performed for non-burnup credit cask
NEI 4	Comment on specific loading procedure: assurance that there is no fresh fuel in the pool during system loading. This procedure would not contribute to preventing a misload of an assembly under any of the three situations identified. While it is acknowledged that this would mitigate the consequences of 1) loading the wrong fuel assembly, it is also recognized that such an occurrence would be more effectively prevented by our recommended	Staff agrees that this procedure would not reduce the likelihood of a misload; however, the draft ISG is not recommending consideration of fresh fuel assemblies for the misload analysis. This is due in part to the results of the NUREG/CR-6955 misload consequence analysis, which demonstrate that a burnup credit cask misloaded with a single fresh, 5.0 weight % fuel assembly will not remain adequately subcritical. Fresh fuel assemblies are also not recommended	Revised Section 5, *Loading Curve and Burnup Verification*, to revise recommended procedures that address the presence of fresh fuel during loading operations.

	administrative procedures. Therefore, imposing a condition that there is no fresh fuel in the pool during system loading would be unnecessary as it does not decrease the likelihood of a misload event. This condition would be more applicable to the consequences of a misload event and would be better addressed in the consideration of the assumptions for the misload analyses, as discussed in comment #1d. From a practical standpoint, we recognize that licensees typically do not load casks while fresh fuel is in the pool, however, there could be an instance when this is necessary.	for the misload analysis given their obvious physical differences from burned assemblies, which staff believes would make prevention of fresh fuel misloads amenable to simple administrative procedures. Staff recognizes that this imposes an operational restriction on cask users, and that, as discussed in Comment NEI 6, is redundant with the "qualitative" burnup verification recommendation.	
NEI 5	Comment on specific loading procedure: verification of the location of high reactivity fuel (i.e., severely underburned fuel) in the spent fuel pool both prior to and after loading. This procedure would not contribute to preventing a misload of an assembly under any of the three situations identified. The intent of this procedure appears to be to prevent 1) loading the wrong fuel assembly; however, it is insufficient to accomplish this objective since it does not verify the actual assemblies to be placed into the casks. The recommend administrative procedure #1 above would be more effective at preventing loading the wrong fuel assembly, since it verifies the assemblies to be loaded into the cask. The recommended administrative procedures would be more efficient, since many spent fuel pools contain hundreds or thousands of fuel	The specific procedure recommended in the draft ISG is intended to reduce the likelihood of loading spent fuel assemblies that are outside the range of parameters considered in the misload analysis. The fuel assemblies that are intended to be addressed by this recommended procedure are "severely underburned," in that they have a higher reactivity than those considered in the single misloaded fuel assembly evaluation recommended in Section 5 of the ISG. Based on the RW-859 database of discharged fuel as of 2002, staff believes that the number of such fuel assemblies in a particular spent fuel pool will be low, and may be zero in many cases. Staff will retain the recommendation to verify the presence of high reactivity fuel assemblies prior to and after loading, and will revise the ISG to clarify what is meant by "severely underburned."	Revise Section 5 and associated Appendix text to clarify "severely underburned."

	assemblies, and therefore it requires fewer resources to verify the assemblies that will be loaded into the casks rather than the assemblies that will not be loaded into the casks.		
NEI 6	Comment on specific loading procedure: qualitative verification that the assembly to be loaded is burned. This procedure would not prevent a misload under any of the three situations identified. A qualitative verification may prevent a fresh fuel assembly from being selected, however this would duplicate the proposed action to ensure fresh fuel is not in the pool during loading. This condition would be more applicable to the consequences of a misload event and would be better addressed in the consideration of the assumptions for the misload analyses. Furthermore, it is unclear how this would be performed. While "visual" implies that the guidance anticipates some physical change, such as color, may be readily verified, "gross measurement" appears to imply that a burn-up measurement is necessary.	Staff agrees that this recommended procedure is redundant with ensuring that fresh fuel is not present during fuel loading. The ISG will be revised as described in response to comment NEI 4.	Revised Section 5 of the ISG and associated text in Appendix A to revise the recommended administrative procedures for preventing fresh fuel from being loaded.
NEI 7	Comment on specific loading procedure: confirmation that an audit of the pool inventory has been performed no more than one year prior to the time of loading. Reliance on a licensee's QA program to ensure configuration control of the pool inventory is sufficient to ensure that storage of fuel in the pool is consistent with the records, and the performance of a misload analysis provides defense-in-depth. Also	Staff agrees that the recommendation for pool audit is potentially duplicative of MC&A requirements in 10 CFR 74. Staff also notes that this recommendation is redundant with the recommendation to verify the presence of high-reactivity fuel both prior to and after loading. However, staff believes some degree of additional verification is necessary prior to shipment of previously loaded systems, which may have been in storage	Revised Section 5 of the ISG and associated text in Appendix A to replace pool audit recommendation with QA audit for already loaded canisters prior to shipment.

	note that 10 CFR Part 74 contains requirements for material control and accounting (MC&A), for which licensees perform inventories of the pool. The condition in the draft guidance could become duplicative, or possibly impose additional conditions that were purposefully avoided in the requirements of 10 CFR Part 74 because they result in undue burdens. Guidance should not create expectations for which there is not a requirement in Part 72 regulations, when there are requirements established by other Parts of the regulations.	for many years. The ISG will be revised to remove the pool audit recommendation and replace it with a QA audit of already loaded canisters prior to shipment.	
NEI 8	Comment on specific loading procedure: quantitative measurement of any fuel assemblies without visible identification numbers. An assembly's identification number should be visible prior to loading into a cask. As a practical matter, all fuel assemblies will have visible identification numbers, and will be verified by existing administrative procedures. If there is a case where the assembly identification number is not visible, then it should not be permitted to be loaded, unless other means were developed and have prior NRC approval. Since this situation is not anticipated and would be a highly unlikely case, we do not believe that guidance needs to accommodate loading fuel without a visible identification number.	Since all fuel assemblies will eventually need to be transported, staff believes an administrative procedure which would provide an option for shipping fuel assemblies without visible identification numbers is necessary. This recommended procedure is burnup measurement in the draft ISG, although other procedures or analyses attempting to address lack of identification will be considered. Note that staff has had informal conversations with some applicants and licensees indicating that this situation may exist at some facilities. This recommended procedure will be maintained in the final ISG.	None.
NEI 9	Comment on specific loading procedure: independent, third party verification of the loading process. Commenter notes that NRC routinely credits licensee QA	Staff recognizes that this recommended procedure goes beyond typical QA requirements. However, in staff's studies related to misload probability, the addition of	Revised Section 5 of the ISG and associated text in Appendix A to clarify that independent verification

	programs for ensuring configuration control for activities such as reactor core loading and spent fuel pool storage loadings, and does not believe that more a more rigorous NRC expectation for cask loading is warranted. As a practical matter, many licensees do include verification of the loading process by a third independent individual. It would not be appropriate for the guidance to impose conditions that exceed NRC QA requirements, and that decisions on whether or not to institute practices that go beyond the NRC's QA requirements is best determined by individual licensees.	such a verification step was determined to result in a significant reduction in misload probability, compared to other procedure steps. Additionally, staff concurs with NEI's assessment that this is already typically done by licensees using dry storage systems. This recommendation will be revised to specify that third party verification should cover the fuel selection process and generation of the fuel move instructions.	should begin with the fuel selection process and generation of the fuel move instructions.
NEI 10	The role of the misload analysis should be redefined as defense-in-depth to the administrative procedures in a risk informed manner. The draft guidance places the administrative procedures in a defense-in-depth role to the misload analyses. Commenter believes the draft guidance has reversed the role of these; i.e. it is the administrative procedures that prevent a misload and the misload analyses are performed as defense-in-depth to ensure the consequences of a highly unlikely human error resulting in a misload would be acceptable.	Staff believes the role of the misload analysis as identified in the ISG is appropriate. The misload analysis demonstrates that even with a severe misload, the storage or transportation system is very likely to remain subcritical in fresh water, and the administrative procedures act as a defense-in-depth measure to reduce the probability of occurrence. This order is especially important given that dry storage misloads have been demonstrated to not be "highly unlikely," as stated in the comment.	None.
NEI 11	The draft guidance permits "…an appropriate administration margin that is not less than 0.02Δk" provided "An adequate justification, that includes the level of rigor in the evaluations and benchmark methods, should accompany the use of any administrative margin that is	Staff believes the appropriate justification for a reduced administrative margin is discussed in some detail in the last paragraph under "Misload Evaluation" in Section 7 of Appendix A, and in much greater detail in the referenced FCSS ISG-10. However, this section of the ISG and its	Revised Section 5 of the ISG and associated Appendix A text to clarify "adequate justification" recommendation related to misload analysis administrative margin.

	less than the normal 0.05Δk". Commenter recommends that the guidance remove the statement on "adequate justification" in favor of generically granting an administrative margin of 0.02Δk for misload analyses. It is not clear what the NRC considers to be an adequate justification, nor how the level of rigor would impact the justification.	associated Appendix A text will be revised to clarify this statement.	
NEI 12	A single misload analysis is desirable as it provides defense-in-depth to the prevention of misloads provided by administrative procedures by ensuring that a single misload would remain subcritical. The assumptions of the misloaded assembly in the draft guidance are overly complex, and the intent and definition of a "severely underburned assembly" to assume for the single misload is not clear. A simpler, more bounding assumption for the single misload would be "the most reactive fresh fuel assembly in the most reactive cask location". Since this is an extremely conservative assumption, it is also recommended that the guidance explicitly state that alternative assumptions for the single misload assembly may be acceptable if justified by consideration of realistic fuel characteristics in the pool and/or administrative procedures.	Staff agrees that the single fresh fuel assembly in the most reactive cask location would be bounding, and would certainly be acceptable as a single misload condition. However, most existing storage or transportation systems will not be able to demonstrate adequate subcriticality under this assumption. The misload characteristics recommended in the ISG are reasonably bounding, and flexible enough to accommodate various capacity systems and fuel populations. Staff will clarify the definition of a severely underburned assembly in the ISG.	Revised Section 5 of the ISG and associated text in Appendix A to clarify the definition of severely underburned.
NEI 13	The assumptions of the multiple misloaded assembly analysis should be developed on a risk informed basis which considers the risk reduction of a multiple misload event through the administrative procedures. The proposed assumptions in the draft guidance	Staff recognizes that the alternative proposed by NEI is simpler, and would be more conservative in certain instances. However, the alternative would be less conservative in some cases, and is dependent upon the specific loading curve.	Revised Section 5 of the ISG and associated text in Appendix A to clarify moderately underburned fuel definition.

	are overly complex, and the intent and definition of a “moderately underburned assembly” to assume for the multiple misload is not clear. It appears that the fuel population analysis would require extensive resources to perform and would be difficult for licensees to verify compliance of fuel loaded into the casks, and difficult for the NRC to perform a review and oversight. Commenter recommends an assumption of “50% of the fuel being misloaded with assemblies that have burn-up reduced by 25%.” This should be sufficiently conservative, and is more straightforward for performing analyses and verifying that implementation by licensees is consistent with the cask licensing basis. It is also recommended that guidance explicitly state that alternative assumptions may be acceptable if justified by consideration of realistic fuel characteristics in the pool and/or administrative procedures.	The recommended multiple misload characteristics in the ISG are intended to be independent of the loading curve, while recognizing that misloaded fuel may be higher or lower than the curve. Staff considers the recommended misload conditions to be reasonably bounding. Staff will, however, revise the definition of moderately underburned fuel in the ISG to be more clear.	
NEI 14	Commenter recommends that burn-up measurements be completely removed from the guidance. Solely relying on the reactor records is justified as they are very accurate, and have a long history of use and acceptance by NRC for reactor operations and spent fuel pool storage. Additionally, in-pool burn-up measurements are difficult to perform, result in worker dose and costs, as well as diverting resources away from activities that are more important to safety.	The burnup measurement recommendation is maintained as an option in the draft ISG, and does not need to be performed if the applicant performs a misload analysis accompanied by additional administrative procedures. The measurement recommendation will be maintained in the ISG as an alternative to misload analysis/admin procedures. This will allow flexibility to applicants if the misload analysis criteria are too restrictive for their specific design, and will allow for future measurement techniques which may make measurement option more appealing.	None.

NEI 15	Guidance that includes more flexibility to use alternative methods for code validation would ensure safety and regulatory compliance, while also ensuring efficient use of industry and NRC resources. The draft guidance endorses the methods for code validation for burn-up credit documented in NUREG/CR-7108 and NUREG/CR-7109; however, it is not evident from the draft guidance that the NRC would accept alternative methods if appropriately justified. Alternative methods for performing the code validation for burn-up credit may become available; in fact, an alternative method currently exists and was published by EPRI in 2011. Industry has expressed interest in using this method for code validation for burn-up credit. We recommend that the guidance be improved to be clear that alternative approaches can be proposed by applicants, and if sufficiently justified, approved by the NRC.	The validation methodology recommended by ISG-8 represents one method that has been reviewed in detail by the staff and found to be acceptable. The ISG is not intended to exclude alternative methodologies, as these would be evaluated on a case-by-case basis.	Revised Introduction to clarify that alternative methodologies will be considered on a case-by-case basis.
NEI 16	Guidance that includes more flexibility to credit additional isotopes beyond those listed in the guidance would ensure safety and regulatory compliance, while also ensuring efficient use of industry and NRC resources. The draft guidance currently limits the actinides and fission products to those listed in Tables A-1 and A-2. While the nuclides permitted are those with the most impact on reactivity, this set does not represent full burn-up credit. Commenter recommends the guidance explicitly state that "Nuclides included as part of burn-up credit for criticality analyses should be	Staff agrees that the ISG should address credit for isotopes beyond those recommended in the guidance. Staff will modify the ISG to state that additional isotopes may be credited, provided the bias and bias uncertainty associated with those isotopes are quantified.	Revised Section 1 of the ISG and associated Appendix A text to clarify that additional isotopes may be credited, provided the bias and bias uncertainty associated with those isotopes are quantified. Additional isotopes will be considered on a case-by-case basis.

	included in the code validation. Nuclides that are not included in the code validation would need to be justified. Assumptions, if demonstrated to be conservative, may be considered appropriate justification."		
NEI 17	Commenter recommends that the guidance explicitly acknowledge the potential for an applicant to take credit for burn-up for BWR fuel, and that consideration of ISG-8 may be useful to applicants developing an approach for NRC review and approval, and that the following be explicitly stated in the guidance: "While this revision to ISG-8 does not specifically provide guidance for taking burn-up credit of BWR fuel, such an approach could be found acceptable if appropriately justified, and should consider the portions of this ISG that are also applicable to BWR fuel." While not specifically requesting the unique aspects of BWR fuel, as they relate to burn-up credit, to be included in ISG-8 at this time, such guidance may be desired in the future.	Staff notes that NRC has initiated research to support guidance on BWR burnup credit for storage and transportation, and that this research will not be completed for 2-3 years. However, the staff agrees that the guidance should acknowledge the potential for applicants to develop BWR burnup credit approaches for NRC to review.	Revised Section 1 of the ISG and associated Appendix A text to state that BWR burnup credit applications will be reviewed on a case-by-case basis.
NEI 18	Commenter recommends that the ISG's use of the main body and Appendix A be improved to better align with the dual uses by 1) NRC staff, and 2) applicants, licensees and CoC holders. Ease of understanding and use of the draft guidance could be improved if these two parts of the document have well defined purposes. This would also eliminate two attributes of the draft guidance that increase its complexity: 1) that some content is duplicated between these two parts of the document, and 2) that some	Staff believes that the current organization of the ISG is sufficient to communicate to staff the concepts necessary for performing reviews of burnup credit criticality analyses for PWR storage and transportation systems. The current organization is consistent with previous revisions of the ISG, and will facilitate future revisions to incorporate new burnup credit methodologies (e.g., BWR burnup credit), as they become available. The ISG and Appendix text will be incorporated directly into the criticality chapters of the spent fuel	None.

	parts of the main body cannot be fully understood without referring to the Appendix.	transportation and storage SRPs.	
NEI 19	Commenter recommends the guidance include a section describing the regulatory basis. Other Interim Staff Guidance documents include this discussion, and it provides clarity and completeness to the overall guidance. The “Regulatory Basis” section, which could be included between the “Introduction” and “Applicability” sections, should cite the applicable regulations for which the guidance is establishing an NRC position. These regulations may include: 71.55(b), 71.55(d)(1), 71.55(e), 72.124(a), and 72.236(c).	Staff agrees with commenter that a Regulatory Basis section should be included in the ISG.	Revised the ISG to include a Regulatory Basis section between the Introduction and Applicability sections.
NEI 20	Commenter notes that other ISG documents include a discussion on the applicability of the ISG to the existing Standard Review Plans in the “Applicability” section. The “Applicability” section should be expanded to explain to which Standard Review Plans (and the specific sections) ISG-8 applies, and how it applies. This may include the following: • SRP NUREG-1536, Section 7.5.5: replaced in its entirety by ISG-8 Revision 3 • SRP NUREG-1567, Section 8.4.5: replaced in its entirety by ISG-8 Revision 3 • SRP NUREG-1617, Section 6.5.8: replaced in its entirety by ISG-8 Revision 3	Staff agrees with the commenter that the guidance should explicitly state which SRPs this ISG is applicable to.	Revised the ISG to include SRP references in a Recommendation section.
NEI 21	Commenter recommends that the guidance include a section listing the acceptable codes and standards, if any. If the NRC intends to accept the standards referenced	ISG-8 is not intended to endorse any specific codes or standards.	None.

	in the draft guidance, then an “Acceptable Codes and Standards” section, which could be included between the “Introduction” and “Applicability” sections, should cite the applicable codes and standards that the draft guidance is endorsing.		
NEI 22	Commenter recommends that the guidance be revised to state that the applicability is also to “undamaged” fuel, and not only “intact” fuel. This is needed to be consistent with the NRC recommendation in ISG-1 Revision 2, page 9. It would also be helpful to include a discussion on the basis why the applicability is not readily extended to “Damaged” fuel, so that applicants will be able to understand the concern, or unique aspects, that must be addressed in a proposed approach for burn-up credit for these conditions of fuel.	Staff agrees that the ISG guidance should also be applicable for undamaged and damaged fuel, per the definitions in ISG-1.	Revised Applicability section of ISG, and associated Appendix A text, to clarify that the guidance is also applicable to undamaged and damaged fuel, provided any additional uncertainties associated with such fuel are addressed in the evaluation.
NEI 23	This ISG will be applicable to any Part 72 Site Specific Storage license that will incorporate burnup credit into their criticality analyses. Site Specific Licenses typically do not have a Certificate of Compliance associated with them. Please change this to “certificate or license conditions”.	Agree.	Revised Section 1 of the ISG, and associated Appendix A text, to state “certificate or license conditions.”
NEI 24	Provide clarification regarding which inputs may be “representative” and which should be bounding. Some of the input items listed here have a 2nd or 3rd order affect on the analysis and thus representative values should be acceptable while others should bound the actual contents of the package.	The last paragraph of Section 2 partially addresses this comment. In general, realistic assumptions and input parameters which only produce small positive increases in k-eff should not be disregarded simply because they are small. Those that are not bounding should be justified in the criticality analysis or tied to specific limits in the certificate or license.	None.
NEI 25	While inputs listed here should be	Staff agrees that not all analysis input	None.

	accounted for in the analysis, not all can be verified for each assembly loaded into the cask. In addition, these parameters are not constant over the life of the fuel assembly or over the axial height of the assembly. Verification of input parameters should be limited to those that are readily available, such as power level.	parameters should be verified for each assembly. Parameters used in the criticality analysis should be selected to bound the population of spent fuel intended to be stored or shipped to the extent practicable. Those that are not bounding should be justified in the criticality analysis. The ISG states that those not selected to be bounding may need to be included in the certificate or license conditions as a loading limitation.	
NEI 26	Commenter recommends that guidance on performing adjustments with regards to control parameters be modified as follows: "The burnup credit results should be adjusted using the bias and bias uncertainty determined for the fuel depletion code, *as adjusted for any trends of significance with respect to any* ~~with regards to different~~ control parameters ~~such as~~ *(these might include* enrichment, burnup, and*/or* cooling time)." NUREG/CR-6811 shows a trend on burnup but does not discuss trends on enrichment or cooling time. Since the two cooling time changes that are important are the Pu-241/Am-241 and Eu-155/Gd-155 decays then a seeking a trend on these ratios may be illuminating.	Staff agrees that enrichment, burnup, and cooling time are not the most appropriate for burnup credit depletion code bias trending analysis.	Revised Section 3 of the ISG to recommend burnup/ enrichment and $^{235}U/^{239}Pu$ ratios as trending parameters for depletion bias.
NEI 27	Clarify whether the "burnup range" corresponds to assembly average burnup or the burnup of a given axial node. The tables should be clarified to state that that these values are the bias uncertainty, and the bias to be used is zero.	ISG will clarify that burnup ranges are assembly average. Also, will revise the tables to report bias and bias uncertainty separately, as appropriate.	Revised Tables 1 and 2 to state that burnup ranges are assembly average values. Also revised Table 1 to state that corresponding bias value

			is zero, and Table 2 to show separate bias and bias uncertainty values.
NEI 28	Consider relaxing the restriction on the use of the pre-determined depletion bias and bias uncertainty to “the same depletion code and cross section library.”, or providing this possibility if appropriately justified. This will allow the use of MCNP, which for the same isotopic content agrees very well with KENO.	Agree with commenter that the restriction here should be for the same depletion code.	Revised Section 3 of the ISG and associated text of Appendix A to clarify that this restriction applies to the depletion code only.
NEI 29	It is unclear what the “similar initial assumptions” means, as these will depend on the limiting conditions expected for the fuel to be loaded in the cask. These assumptions are different than using the actual conditions for a chemical assay. If the applicant is using the values in Table 1, then initial assumptions of the chemical assays would not be relevant. It is unclear what the “code modeling options” means, as NUREG/CR-7108 does not provide input decks from which the code modeling options used could be determined. Is this intended to apply to ENDF/B-V where it is possible to use NITAWL rather than CENTRM and this will produce different results?	Staff agrees that the initial assumptions and code modeling options used in NUREG/CR-7108 and -7109 are not readily available to applicants and licensees. ORNL modeled a large number of depletion and criticality cases, including a number of sensitivity studies, which envelop a relatively large range of initial assumptions and modeling options. Staff believes that while there are several modeling options that could have an effect on the applicability of the given bias and bias uncertainty values, it is more important that they are correct for the situation being modeled than that they are the same as what was used in the NUREG/CRs.	Revised Sections 3 and 4 of the ISG, and associated text in Appendix A, to revise the applicability clauses regarding initial assumptions and code modeling options to refer instead to “appropriate initial assumptions and input parameters, as described in Appendix A.”
NEI 30	Biases and uncertainties should not be directly combined in determining the final k-eff. Biases are added directly to the calculated k-eff, while uncertainties are statistically combined with each other before being added to the calculated k-eff. Correct the references to combining biases	Staff agrees that biases and bias uncertainties should typically be reported and treated separately. This has been done with the depletion bias and bias uncertainty as described in the response to NEI 27. However, the uncertainty in k-eff due to uncertainty in the cross section data	Revised Section 6 of Appendix A to clarify the use of the combined bias and bias uncertainty for k-eff determination.

	and uncertainties, and clarify the values by separating them into their constituent parts.	reported in NUREG/CR-7109 is intended to be used as a bias. This is because there is no critical experiment information to determine a traditional k-eff bias, and the uncertainty determined by ORNL is an indication of how large the bias could be, based on cross section uncertainties.	
NEI 31	Commenter requests that the referenced standard not be quoted since the use of the word "shall" in guidance conveys a requirement, and would not be appropriate. In cases where references use words such as "shall", it is recommended that the guidance summarize the reference or cite it without direct quotation. In this particular use, we recommend the following citation of the standard "ANSI/ANS 8.1 provides an acceptable method for establishing the bias by correlating the results of critical and exponential experiments with results obtained for these same systems by the calculational method being verified. Other methods may be used if appropriately justified."	The use of "shall" in this instance is intended to imply that the action is necessary to be incompliance with ANSI/ANS 8.1, not that it is necessary to be in compliance with the ISG. Also note that this explanatory text appears in the guidance, and does not convey a recommendation to the staff.	None.
NEI 32	The use of "must" conveys that no alternative can be proposed, and in certain context could effectively establish a requirement. Many of these uses of "must" in the draft guidance are in conditional statements; however, there may be valid alternatives to the absolute condition being imposed by these statements. In these cases, the use of "must" eliminates the applicant's ability to propose such an alternative. For conditional statements where "must" is used, either replace "must"	Staff has evaluated the use of "must" in the instances noted by the commenter. In most cases, staff believes the use of "must" to be appropriate. For instance, on page A-3, the use of "must" is associated with the conditions for directly using the bias and bias uncertainty numbers developed in the ORNL NUREG/CRs. There are alternative validation methodologies, but in order to use the numbers directly, the conditions cited in this section must be done.	Revised Section 5 of Appendix A (page A-19) to change several instances of "must" to "should."

	with a softer conditional statement, such as “should”; or follow such conditions with a statement that alternative approaches may be acceptable if appropriately justified.		
Lancaster 1	The restriction to "Intact fuel" originated in the "Topical Report on Actinide-Only Bumup Credit for PWR Spent Nuclear Fuel Packages.” The intent of this restriction was to exclude significant fuel movement. ISG-1 calls fuel with pinhole and hairline defects in the clad as not intact. From a criticality point of view these defects are insignificant. If ISG-1 definitions are to be used then "intact fuel" should be changed to "fuel that is not grossly breached."	See response to NEI 22.	None.
Lancaster 2	NUREG/CR-6811 shows a trend on burnup but does not discuss trends on enrichment or cooling time. Since the two cooling time changes that are important are the Pu-241/Am-241 and Eu-155/Gd-155 decays then a seeking a trend on these ratios may be illuminating. Commenter recommends that the enrichment and cooling time be eliminated from the referenced sentence. The remaining sentence would still say "with regards to different control parameters such as burnup." This sentence then suggests that the applicant carefully review the data and underscores the important parameter, bumup.	See response to NEI 26.	None.
Lancaster 3	The tables need to be clear that the values are the uncertainty in the bias and the bias to be used is zero. The uncertainty can be	See response to NEI 27.	None.

	statistically combined with other uncertainties. No statement on statistical combination is necessary unless the NRC is not allowing combination of this uncertainty with other uncertainties.		
Lancaster 4	The depletion bias results depend on the cross section library. They depend on the depletion code to a lesser extent. They should not depend much on the final criticality code (KENO vs MCNP). Consider relaxing the restriction to "the same *depletion* code and cross section library." This will allow the use of MCNP which for the same isotopic content agrees very well with KENO. Alternatively, allow for some sample cross checking between KENO and MCNP to prove acceptability.	See response to NEI 28.	None.
Lancaster 5	It is unclear what "similar initial assumptions" means. The depletion assumptions for cask analysis will depend on the limiting conditions expected for the fuel to be loaded in the cask. These assumptions are different than using the actual conditions for a chemical assay. Remove "initial assumptions and."	See response to NEI 29.	None.
Lancaster 6	The restriction on code modeling options is also not clear. NUREG/CR-7108 does not provide input decks. However, for ENDF/B-V it is possible to use NITWAL rather than CENTRM and this will produce different results. Commenter recommends that Table 2 specify CENTRM so this issue is removed. Also, recommend removing this restriction or providing more details about what it means.	See response to NEI 29.	None.
Lancaster	The appendix is clearer on what similarity to	Acceptable ranges for these example	None.

7	the GBC-32 means but does not actually give an acceptable range for the H/X or EALF. Commenter recommends that the specific range for these parameters be determined and that the range be included in the Appendix where this is discussed.	system parameters, and others, would be determined by the applicant's model of the GBC-32, in comparison with the same fuel in their application system. This would be similar to the manner in which critical experiments are compared to an application system to determine their applicability. Alternatively, the applicant may use sensitivity and uncertainty analysis tools to compare the systems, as is described in Section 2 of Appendix A to the ISG.	
Lancaster 8	Use of "combined bias and bias uncertainty" should be avoided. Uncertainties are statistically combined but biases are added. Recommend changing text in Section 4 to: "1.5% of the worth of the minor actinides and fission products conservatively covers the bias due to these isotopes. Due to the conservatism in this value no additional uncertainty in the bias needs to be applied."	See response to NEI 30. Also, staff agrees with the proposed text.	Revised Section 4 of the ISG, and associated text in Appendix A, to incorporate suggested language.
Lancaster 9	The combined minor actinide and fission product worth for high bumups is close to 0.1 in k-eff. For example in one case it was calculated as 0.11 in k-eff. The range of data used in NUREG/CR-7109 can justify a higher limit for the range of applicability. Recommend raising this to 0.13 to give more margin for various designs.	The 0.1 credited minor actinide and fission product worth restriction is based on the fact that none of the sensitivity studies performed to determine the applicable k-eff bias showed worths greater than that value (with few above 0.08). Note that this restriction is given to one significant figure, such that a calculated worth of 0.11 would still be acceptable.	None.
Lancaster 10	The recommended bias of 1.5% of the worth of minor actinides and fission products depends mainly on the cross section library and should be the same for codes other than SCALE. Proof of this for other codes is not possible yet but the	Staff agrees that the 1.5% recommendation should be applicable to other industry standard codes, provided the same cross section data is used and there is some demonstration of applicability – possibly via a minor actinide and fission product worth	Revised Section 4 of the ISG, and associated text in Appendix A, to allow for applicants to demonstrate applicability of the 1.5% criterion for other code

	factor of 2 increase in the bias should not be needed. Comparison of fission product and minor actinide worth between SCALE and code of choice could be used to confirm applicability to other codes. For this comparison a benchmark should be set up showing the isotopic worths in SCALE and then other codes could be compared to this benchmark. Recommend removing the requirement to use the SCALE system.	comparison. However, the recommendation for a benchmark where the worths are provided for the GBC-32 system requires more work, and will not be available for this revision of the ISG. Staff will consider performing this work to be included in follow-on guidance.	systems.
Lancaster 11	Page A-2 discusses "intact" fuel. Here, the discussion seems to follow the original intent on "intact" fuel but is inconsistent with the ISG1 definition. The concern is over Reconstituted, disassembled or grossly damaged fuel. The term "intact" needs to be replaced.	See response to NEI 22.	None.
Lancaster 12	Page A-3 middle paragraph provides the limits discussed in Comments 4-7. Please update to be consistent with responses to Comments 4-7.	Agree.	Revised Page A-3 of Appendix A of the ISG to be consistent with responses to earlier comments.
Lancaster 13	Top of Page A-12. It seems to suggest that the applicant calculate end effects. The applicant should not be required to calculate the uniform burnup k at burnups where the end effect is a positive contribution to reactivity. The applicant will have to assure the more limiting burnup profile is used in the burnup range of transition but calculation with the non-limiting profile should not be required. "demonstrate that the Δk value(s)" should be removed.	Agree.	Revised Section 4 (Page A-12) of Appendix A of the ISG to remove the phrase “demonstrate that the Δk value(s)"
Lancaster 14	It is conservative to assume fuel is not blanketed and use the limiting axial profiles.	Agree.	Revised Section 4 (Page A-13) of Appendix A of the

	The current writing of the 2nd paragraph on page A-13 may lead one to believe there is not a solution for blanketed fuel. The rest of the paragraph starting with: "While the database included some assemblies with axial blankets" should be deleted or rewritten.		ISG to rewrite the paragraph to indicate that it is non-conservative to model blanketed fuel with non-blanketed axial distributions.
Lancaster 15	The statement on Page A-14 regarding "over 1000 nuclides" is not supported. From a spectrum point of view, SCALE does not support more than 388 isotopes. Precursors to the 28 nuclides credited do not require 1000 isotopes. There is no documentation that indicates using fewer isotopes produce poorer results. This paragraph is on a non-issue and should be deleted unless there is a real issue with a code system that could be used.	Agree in part. This paragraph conveys important information to a reviewer regarding the operation of depletion codes.	Revised Section 4 (Page A-14) to replace "1000" isotopes with "a large number of" isotopes.
Lancaster 16	On Page A-14, the X-Y plane at each segment is used for power reactors but since for PWR cask analysis axial variation of enrichment is rarely credited this discussion is not relevant to cask criticality. Most cask criticality calculations use a uniform axial temperature assumption. More complicated assumptions would be hard to justify for a cask. This paragraph has no value and should be deleted.	This statement in Appendix A is intended only to address axial profile modeling, not axial variation of fuel design or irradiation parameters.	Revised Section 4 (Page A-15) to clarify the statement regarding axial profile modeling.
Lancaster 17	The suggestion on page A-15 that CASMO or HELIOS would not be adequate for analysis of depletion for a cask is dismaying. The presence of some lumped fission products does not disqualify a depletion code. Decomposing a lumped fission product to some of the 28 allowed isotopes would be surprising. If that were	This paragraph was not intended to imply that these codes cannot be used for burnup credit depletion analyses.	Revised Section 4 (Page A-15) of Appendix A of the ISG to remove the implication that these specific codes are not adequate for depletion analyses.

	attempted the regulator would of course have to be very careful. This paragraph should be deleted.		
Lancaster 18	The paragraph on Page A-15 regarding 1D approaches seems to show little acceptance that the supercell approach had been worked out. It is doubtful that anyone will use a 1D approach in the future but it is disrespectful to not recognize that our elders had worked out these issues. This paragraph should be deleted.	The intent of this paragraph was not to disparage 1D depletion codes, but to point out to reviewers that there are additional approximations related to their use, when compared to 2D codes.	Revised Section 4 (Page A-15) of Appendix A of the ISG to remove the implication that 1D codes are unacceptable.
Lancaster 19	The final paragraph on Page A-15 seems to be making an issue out of reactor operating history. The previous sections have dealt with these issues. Both the time and space meshing used in the analysis should be converged. This is tested by decreasing the time and space mesh until the changes in the results are consistent with the desired accuracy. This paragraph can be deleted without loss.	This paragraph is not intended to make an issue out of reactor operating history, but merely to point out that this history is important to appropriately model in the depletion analysis. Also, this paragraph is pointing out that the number of time steps in which the burnup-dependent cross sections are updated is an important parameter to review.	None.
Lancaster 20	Page A-16: "A uniform loading of SNF at a specified assembly average burnup..." There is no reason a uniform loading is required. The NRC has already approved a burnup credit design with zoned loading. This sentence should be removed.	Agree.	Revised Section 4 (Page A-16) of Appendix A of the ISG to remove the recommendation for uniform loading.
Lancaster 21	Page A-16: "18-20 uniform axial regions." Modeling fuel with 24 nodes is very common. Increase 20 to 24.	Agree.	Revised Section 4 (Page A-16) of Appendix A of the ISG to increase 20 to 24.
Lancaster 22	For the first paragraph on Page A-17 regarding source convergence, provide a reference so the reader will know what type of source assumption can cause troubles.	Agree.	Revised Section 4 (Page A-17) of Appendix A of the ISG to clarify that starting particles in the more reactive ends of the fuel may improve convergence

			to the correct k-eff.
Lancaster 23	On the first paragraph on Page A-18 regarding number of RCA samples: Only two samples contain all 28 isotopes. This would seem to be a problem with regards to the second sentence. Cs-133 is based on only 7 samples. Ag-109 has only 14 samples, Ru-101 and Mo-95 have only 15 samples, and Rh-103 has only 16 samples. The comment that the sample size should be 30 is clearly not the case for the basis for the recommended biases. Recommend deleting this paragraph.	Although at least 30 samples are desirable, there are appropriate methods for dealing with smaller sample sizes. The use of smaller sample sizes should be accompanied by additional statistical analyses and methods to support their use, as was done with the samples used in NUREG/CR-7108.	Revised Section 5 (Page A-18) of Appendix A of the ISG to include a statement indicating that smaller sample sizes should be accompanied by additional statistical analyses and methods to support their use.
Lancaster 24	The second paragraph on page A-25 points out that there could be compensating errors that are not able to be found by the integral approach. This is true but misses the fact that there may be compensating errors in our standard approach. We do critical experiments that use cross sections for a large number of isotopes. We have errors in our U-235 cross sections which are compensated for by errors in our U-238 cross sections. We look for these errors by our trend analysis but we certainty do not get rid of the compensating errors. In the integral approach we may indeed have errors in the isotopic content. We try to understand these errors through chemical assays. Unfortunately, the chemical assay data is much more uncertain than our measurement of core reactivity so all we can do with the chemical assays is see gross errors. We feel good about our criticality validation if it is representative of the critical condition of concern. The CRCs	This section of the Appendix is intended mainly to point out to the reviewer the many issues associated with the use of CRCs for integral depletion and criticality benchmarking. The ISG represents one way to approach depletion and criticality benchmarking. Other approaches will be considered on a case-by-case basis, as indicated by the last sentence in this section.	None.

	have a high c_k values. The c_k values for the CRCs are higher than the c_k values for the critical experiments that will be used for our validation. Should we not worry more about compensating errors in critical experiments than the compensating error between our depletion and criticality codes? The third paragraph raises concerns that were addressed in the TSUNAMI analysis. The one valid complaint is the complexity of the modeling. Missing is the main reason the EPRI work was done: the maximum core average bumup is 33 GWd/MTU. This core average burnup is a volume weighted value and the importance weighted burnup would be less. It is recommended that this section be reduced to: "'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…"		
Anton 1	The ISG does not specifically address how the maximum k-eff is to be calculated. The referenced recently published NUREGs list the following equation (based on ANSI/ANS-8.27): $k_p + \Delta k_p + \beta_i + \Delta k_i + \beta + \Delta k_\beta + \Delta k_x + \Delta k_m < k_{limit}$ This equation adds uncertainties	Agree. See response to NEI 27. Staff will also revise the ISG to give a separate β_i, Δk_i, and Δk_x, and to indicate that Δk_i may be statistically combined with other independent uncertainties.	Revised Sections 3 and 4 of the ISG, and associated text in Appendix A, to clarify the use of β_i, Δk_i, and Δk_x.

	arithmetically. The ANSI standard clarifies that independent uncertainties may be combined statistically, but the NUREG is silent on this issue. The ISG should clarify that a statistical combination of independent uncertainties is acceptable.		
Anton 2	Page 1, "This ISG revision also includes an increase in the assembly average burnup recommended for burnup credit." This sentence should be revised to clarify that the upper burnup limit is increased. As written, the sentence could be interpreted to state that there is a recommended burnup value for burnup credit, which is now higher than before.	Agree.	Revised the Introduction to the ISG, and associated Appendix A text, to clarify that the recommended assembly average burnup is a maximum.
Anton 3	Page 2: ISG-1 Rev. 2 distinguishes between intact and undamaged fuel, where undamaged fuel may have certain defects as long as the important performance functions of the fuel is not impaired. If it is in fact the intent to limit burnup credit to intact fuel, then this should be discussed and justified. Otherwise, "intact" should be changed to "undamaged".	See response to NEI 22.	None.
Anton 4	Page 2, "accurate representation of the physics in the system." It should be clarified that if models, assumptions and inputs appropriately consider all phenomena discussed in the ISG, then the accuracy requirement is satisfied. Without such a clarification, an applicant would be unable to demonstrate that this accuracy requirement is fulfilled.	The bulleted list and paragraphs which immediately follow this sentence describe what the reviewers should evaluate regarding representation of the “physics in the system,” with more detail provided in Appendix A.	Revised Section 2 of the ISG to point to parameters given in the ISG and Appendix A regarding accurate physics representation.
Anton 5	Page 3, Paragraph starting "YAEC-1937 ..." Recommend removing "for each burnup range" after "proposed contents". It implies	Axial profiles are routinely generated for different burnup ranges on the same loading curve.	None.

	that axial profiles are established for more than one burnup range, which may or may not be the case.		
Anton 6	Page 4, Section starting "In lieu of an explicit benchmarking ..." Traditionally, one additional purpose of benchmarking was to qualify the individual or organization performing the calculations. How is this achieved now? Note that this aspect may be specifically important for depletion codes, which are more specialized and not as widely used as Monte Carlo criticality codes.	For k-eff determination, the applicant still must perform a validation of the code for the major actinides, which represent the majority of the decrease in k-eff with burnup. Staff believes that this is sufficient to qualify the individual performing k-eff calculations. For depletion analyses, staff agrees with the commenter. For applicants that choose the route of using the same code and cross section libraries with the ORNL-determined depletion bias and bias uncertainty, reviewers will have to ensure that the applicant has used the code properly, with appropriate initial assumptions and code modeling options.	Revised Section 3 of the ISG, and associated text in Appendix A, to reinforce that the reviewer needs to ensure that appropriate initial assumptions and code modeling options are used for the depletion and criticality analyses.
Anton 7	Regarding Page 6 on criticality bias: since this bias is for a cross section uncertainty, and the relative reactivity effect should be the same for all high quality criticality codes using those cross sections, it is not clear why an increase is necessary. Further, the increase by factor 2 does not appear to have a solid basis, making its justification, other than referencing the ISG, difficult or impossible. Finally, given industries request for a code-independent solution, and the considerable effort that went into developing the NUREGs, it is not clear why this approach was taken. The ISG should therefore either endorse the 1.5% for all high quality codes using ENDF/B-V, VI, or VII cross sections; or provide for an easy verification method that other codes results	See response to Lancaster 10.	None.

	are equivalent to SCALE so they can use the 1.5%.		
Anton 8	Regarding model assumptions, experience has shown that the attempt to bound large fuel populations can result in extremely conservative assumptions and results. More site specific evaluations, using site specific axial profiles, core operating conditions, burnable poison usage, fuel inventories for misloading evaluations, etc., may result in more favorable loading curves. The ISG does not specifically exclude such site specific evaluations. However, the discussions on axial burnup profiles and misload evaluations seem to focus on large fuel populations. The ISG should clarify that site specific calculations and loading curves are permissible.	Agree.	Revised Sections 2 and 5 of the ISG, and associated text in Appendix A, to clarify that site-specific calculations and loading curves are acceptable.
Anton 9	The bulleted list on Page 7 regarding misload analyses should be expanded to include the misload analyses expected on poison rods, burnable absorber and control rods discussed later in that section. It should also be clarified if single or multiple misloads are to be considered for those conditions.	Staff does not have a position on non-fuel absorber misloads at this time; as such systems have not yet been submitted or reviewed. Applicants that wish to credit non-fuel absorbers in a burnup credit criticality analysis should justify their misload analysis assumptions, and such analyses will be considered on a case-by-case basis.	None.
Anton 10	Regarding Page 8, "assurance that there is no fresh fuel in the pool during system loading": given the fact that fresh assemblies can be easily identified, the requirement seems unnecessary, and also operationally impractical.	See response to NEI 4.	None.
Anton 11	Regarding Page 8, "minimum required soluble boron concentration in pool water during loading and unloading:" it is unclear what the basis for the determination of the	Agree.	Revised Section 5 of the ISG, and associated text in Appendix A, to clarify that the soluble boron

	minimum soluble boron requirement is.		recommendation for loading and unloading is a defense-in-depth measure intended to offset the reactivity insertion caused by a potential misload.
Anton 12	Regarding Page A-12, Horizontal Burnup Profiles: the discussion seems contradictory. It states "In large rail casks, the probability that underburned quadrants of multiple fuel assemblies will be oriented in such a way as to have a substantial impact of k-eff is not expected to be significant," but then requests a bias for the effect to be applied.	Although not expected to be significant, the possibility of an increase in system reactivity due to orientation of underburned fuel assembly quadrants should still be evaluated and considered as part of the final calculated k-eff. This statement in Appendix A is intended to inform reviewers that they should not expect a large increase in k-eff due to horizontal burnup profile.	None.
Anton 13	Regarding the first paragraph of the Appendix A section titled, “Depletion Analysis Computational Model:” the depletion code needs to be validated using the approach documented in NUREG/CR-7108, and isotopic correction factors or bias and bias uncertainty are derived from this. The number of isotopes that are tracked in the code appears irrelevant, since any possible shortcomings of the code would be captured by the benchmarking, and only isotopes qualified through the benchmarking are used. Further, even without benchmarking, it is not clear how the number of isotopes can be an objective indication of the quality of the code.	See response to Lancaster 15.	None.
Anton 14	Regarding the third paragraph of the Appendix A section titled, “Depletion Analysis Computational Model:” after determination of isotopic correction factors or bias and bias uncertainty, the question	See response to Lancaster 17. Also, the text of this section is not meant to imply that all depletion codes using lumped fission products are not suitable for burnup credit, only that the use of such codes should be	Revised Section 4 of Appendix A to modify discussion of lumped fission products.

	whether or not a code uses lumped fission products appears irrelevant, since again any shortcomings introduced by those lumped fission products would be captured in the isotopic correction factors or bias and bias uncertainty.	accompanied by additional explanation of the lumping methodology and the methodology for determining specific nuclide concentrations.	
Anton 15	The first sentence in Paragraph 4 of the Appendix A section titled, “Depletion Analysis Computational Model,” appears questionable, and seems to be based on a very narrow definition of "accurate". Two-dimensional depletion codes have been used successfully in the industry for a long time.	The intent of this paragraph is not to imply that 2-D codes are inaccurate, but to indicate that, if they were available, 3-D codes would be preferable for their ability to model axial variation in depletion parameters.	Revised Section 4 of Appendix A to clarify discussion of 2-D vs. 3-D depletion codes.
Anton 16	Overall, the section on Depletion Analysis Computational Model seems to present preferences of one code over others based on qualitative and subjective judgment. However, instead, the qualification of a depletion code should be based on the proposed benchmarking outlined in NUREG/CR-7108. This section should therefore be removed, or at a minimum reduced to the essential content.	This section of Appendix A is not intended to present preferences of one code versus another, but merely to point out to a reviewer what types of codes he or she might expect to see, and the relative advantages/ disadvantages of each.	Revised Section 4 of Appendix A to clarify intent of the paragraphs discussing dimensional aspects of depletion codes.
Anton 17	Regarding Page A-16, "A uniform loading of SNF at a specified assembly-average burnup, initial enrichment, and cooling time should be used for each cask analysis:" it is not clear why only uniform loading should be used. It may be beneficial to qualify certain locations for assemblies of different burnups or cooling times, to increase the overall population of fuel that can be loaded.	See response to Lancaster 20.	None.