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

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1 INTRODUCTION

The concept of taking credit for the reduction in reactivity due to irradiation of nuclear fuel (i.e., fuel burnup) is commonly referred to as burnup credit. The reduction in reactivity that occurs with fuel burnup is due to the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides (non-fissile actinides and fission products. Historically, criticality safety analyses for transport and dry cask storage of spent nuclear fuel (SNF) assumed the fuel contents to be unirradiated (i.e., "fresh" fuel) compositions. In July 1999, the U.S. Nuclear Regulatory Commission (NRC) Spent Fuel Project Office issued Interim Staff Guidance 8 Revision 1 (ISG8R1) to provide recommendations for the use of burnup credit in storage and transport of pressurized water reactor (PWR) spent fuel.[1] These recommendations were subsequently included in the Standard Review Plan for transportation cask and dry storage cask facilities.[2,3] More recently, Revision 2 of ISG8 has been prepared . The purpose of this report is to discuss this latest revision to ISG8 together with the technical basis for each recommendation. The bases for making select revisions to the recommendations of Ref. 1–3 were initially provided in Ref. 4, but have been further documented and enhanced in the list of reports and papers published as part of the research program directed by the NRC Office of Nuclear Regulatory Research (see attached bibliography of Section 10). Published information from the bibliography of Section 10 and other sources provide the basis for the summary technical information and references to help identify and assess the applicant's treatment of important issues.

2 GENERAL APPROACH IN SAFETY ANALYSIS

The assumption of unirradiated fuel at maximum initial enrichment provides a relatively straightforward process for the criticality safety analysis of a storage or transportation cask. Figure 1 provides a schematic interpretation of the steps involved in the criticality safety analysis and loading implementation with the fresh fuel assumption. Similarly, Fig. 2 provides an illustrative schematic for a burnup credit safety analysis and loading implementation. In comparison to the fresh fuel assumption, there is additional information and/or assumptions needed for input to the analysis, additional analyses to obtain the SNF compositions, additional validation efforts for the depletion and decay software, enhanced validation to address the additional nuclides in the criticality analyses, and the verification and pre-shipment measurement to be made prior to loading the cask.

The increased need for technical information on the fuel, the added complexity of the computational modeling and analyses, and the loading verification process all contribute to added complexity in the safety analysis. However, the use of burnup credit provides an additional degree of freedom to the cask licensee and the increased capacities and higher limits on allowed initial enrichments are objectives that motivate the applicant to address the added complexities associated with taking reactivity credit for fuel burnup. resulting from the burnup credit approach.

The important product from a burnup credit safety analysis is the cask loading curve, showing the minimum burnup required for loading as a function of initial enrichment. With an assumed uniform cask loading of SNF, the effective neutron multiplication factor (k_{eff}) will increase with higher initial enrichments, decrease with increases in burnup, and decrease with increased cooling time from 1 y to approximately 100 y. Information that will need to be considered in specifying the technical limits for acceptable loading include: 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 set of loading curves can be generated to define the boundaries between acceptable and unacceptable SNF specifications for cask loading.

The recommendations in Revision 2 of ISG8 include:

1. general information on limits for the licensing basis,
2. guidance on code validation,
3. guidance on licensing-basis model assumptions,
4. guidance on preparation of loading curves,
5. the process for assigning a burnup loading value to an assembly, and
6. the benefit derived in demonstrating any additional reactivity margin beyond that which can be substantiated through the validation process.

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

The six recommendations listed above were developed with intact fuel as the basis. An extension to damaged fuel 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 (or parts thereof) have been adequately addressed. In particular, an appropriate model that bounds the uncertainties associated with the allowed fuel inventory and fuel configuration in the cask must be applied. Such a model should include the selection of appropriate burnup distributions and any potential rearrangement of the damaged fuel during normal and accident conditions. The applicant should also strive to apply each of the recommendations provided in ISG8R2 and discuss or justify any exceptions taken due to the nature of the fuel (e.g., the use of the recommended axial profile database may not be appropriate).

The remainder of this report discusses each of the areas and the associated recommendations (repeated in the body of this report in italics) and provides technical information and/or references that should be considered in the review of the Safety Analysis Report (SAR) against the recommendations of ISG8R2.

3 LIMITS FOR LICENSING BASIS (RECOMMENDATION 1 OF ISG8R2)

Available data supports allowance for burnup credit where the licensing safety analysis is based on actinide compositions associated with UO_2 *fuel irradiated in a PWR to an assembly-average burnup value up to 50 GWd/MTU and cooled out-of-reactor for a time period between 1 and 40 y. The range of available measured assay data for irradiated* UO_2 *fuel indicates that an extension of the licensing basis beyond 5.0 wt % is not warranted. Even within this range of parameters, the reviewer needs to exercise care in assessing whether the analytic methods and assumptions used are appropriate, especially near the ends of the range. Use of actinide 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 adequately extend the isotopic validation and quantify or bound the bias and uncertainty.*

Actinides of Importance. Several studies have been performed to identify the nuclides which have the most significant effect on the calculated value of k_{eff} as a function of burnup and cooling time (e.g., Refs. 5–7). Figure 3 provides the results of one study[7] which performed a relative ranking based on the fraction of total absorptions for each actinide (which has been demonstrated to be directly related to the relative impact on k_{eff}). The relative worth of the nuclides will vary somewhat with fuel design, initial enrichment, and cooling time, but the important actinides (fissile nuclides and select non-fissile absorbers) remain the same and have been substantiated by numerous independent studies. Table 1 lists a recommended set of actinides that may be considered for inclusion in the calculation of the cask k_{eff} value. These nuclides have the largest impact on k_{eff} and, with the possible exception of ^{241}Am (see Section 4, *Prediction of* k_{eff}), there is a sufficient quantity of applicable experimental data available for validation of the analysis methods.[8] Accurate prediction of the concentrations for the actinides of Table 1 requires that the depletion and decay calculations include nuclides beyond those listed in the table. Additional actinides 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 (see Section 5). To accurately predict the fission product margin (see Section 8), explicit representation of the important fission product transmutation and decay chains is needed to obtain the individual fission product compositions.

Table 1 Recommended set of actinides in SNF criticality calculations

^{235}U	^{238}U	^{238}Pu	^{239}Pu
^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am

Burnup and Enrichment Limits. Figure 4 and Table 2 show that the range of existing radiochemical data that are readily available for validation extends up to 47.3 GWd/MTU and 4.1 wt % initial enrichment. Risk-informed technical judgement indicates that trends in the calculational bias and uncertainty derived from this database can be extended for use with SNF having initial enrichments up to 5.0 wt % and average assembly burnups limited to 50 GWd/MTU (local burnups can be higher).[9] Fuel with an average assembly burnup greater than 50 GWd/MTU can be loaded into a burnup credit cask but, based on the limited assay data available for validation, credit should only be taken for the reactivity reduction up to 50 GWd/MTU.

Cooling Time. Figure 5 illustrates the expected reactivity behavior for SNF in a hypothetical 32-element General Burnup Credit (GBC-32) cask assuming use of major actinide concentrations in the calculation of k_{eff}. The fact that the reactivity begins to rise around 100 y after discharge means that 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 y is about the same as that of fuel cooled to 200 y. The low-probability that fuel in a storage or transport cask would remain in place for more than 200 y led to the recommended limiting cooling time criterion of 40 y (i.e., no credit for cooling time beyond 40 y should be taken). The reviewer should note that approval of a cooling time longer than 5 years for burnup credit in dry storage or transportation casks does not automatically guarantee acceptance for disposal without repackaging. Reference 10 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 were set based on the characteristics of SNF discharged to date, the parameter space considered in the predominance of technical investigations, and the experimental data available to support development of a calculational bias and uncertainty. As indicated, a safety analysis that uses parameter values outside those recommended by the interim staff guidance will need to (1) demonstrate that the measurement or experimental data necessary for proper code validation have been included, and/or (2) provide adequate justification that the analysis assumptions or the associated bias and uncertainty have been established in such a fashion as to bound the potential impacts of limited measurement or experimental data.

4 CODE VALIDATION (RECOMMENDATION 2 OF ISG8R2)

The computational methodologies used for predicting the actinide compositions and determining the neutron multiplication factor (k-effective) should be properly validated. Bias and uncertainties associated with predicting the actinide compositions should be determined from benchmarks of applicable fuel assay measurements. Bias and uncertainties associated with the calculation of k-effective should be derived from benchmark experiments that closely represent the important features of the cask design and spent fuel contents. The particular set of nuclides used to determine the k-effective value should be limited to that established in the validation process. The licensing basis safety analysis should utilize bias and uncertainty values that can be justified as bounding based on the quantity and quality of the experimental data. Particular consideration should be given to bias uncertainties arising from the lack of critical experiments that are highly prototypical of spent fuel in a cask

Sources of Uncertainty. Validation is the process by which one demonstrates that the codes and associated data do indeed predict reality. As used in criticality safety, the validation process should include an estimate of the bias and uncertainty associated with using the codes and data for a particular application. For burnup credit applications, the potential sources of uncertainty are numerous. Discrepancies between radiochemical assay data of SNF and computational predictions can arise due to uncertainties in: cross-section and decay data, knowledge of the specific irradiation history and fuel assembly design/use, uncertainty in the recorded burnup (both local and assembly-average), measurement error/uncertainty, and numerical/modeling approximations. Similarly, discrepancies between critical experiment measurement of k_{eff} and code predictions of k_{eff} are typically due to data, modeling, or measurement uncertainties. Beyond these uncertainties is the more difficult one to quantify: uncertainty that may be caused by use of experimental data that does not adequately represent the content and configuration of SNF in a cask. Thus, care is needed to understand and demonstrate the basis for similarity between the measured data used for the validation and the system of interest (i.e., the particular SNF cask design).

SNF Compositions. The credit for burnup is limited to 50 GWd/MTU because the assay data (e.g., Fig. 4 and Table 2) are not available to support development of a bias and uncertainty beyond this burnup without unwarranted extrapolation. From Fig. 4 and Table 2, it can be seen that the primary source of readily-available assay data in the regime above 4.0 wt % and 40 GWd/MTU is from the Takahama PWR in Japan. Work reported in Ref. 9 has demonstrated that the standard deviation of the calculated-to-experimental nuclide ratios for the Takahama data are comparable to those observed for previous lower enrichment and lower burnup assay data. A more definitive analysis of the uncertainties was obtained using a novel technique (see Section 5.1.3 of Ref. 9) that involved using concentrations from the measured actinide assay data directly in a criticality calculation to obtain a k_{eff} value that could be compared with that obtained using predicted nuclide concentrations for the same set of actinides. The difference in the k_{eff} values (Δk) is a direct measure of the bias due to the depletion/decay calculation, and the spread in the Δk values for multiple samples is a measure of the uncertainty. The results for all available assay data (major U and Pu nuclides), plotted in Figs. 6–7 (differences expressed in units of percent reactivity, $\Delta k/k$, where the reference k value is the one associated with the assay inventory), indicate no observed increase in the uncertainty with increasing burnup (Fig. 6) or initial enrichment (Fig. 7).

Figures 6 and 7 both show a similar negative bias trend with increasing burnup and enrichment, as shown by the slope of the linear regression fit of the data. The bias is observed to be relatively small and well behaved. However, the key result of these plots is that they show the uncertainty (determined from the variance of the data) is very uniform over the range of the data—for burnup values up to 47.3 GWd/MTU and enrichments up to 4.1 wt % ^{235}U. An independent analysis of uncertainty using different techniques (but based on the same nuclide validation results) shows similar results (see Sections 5.1.2 and 5.1.4 of Ref. 9). These findings are consistent with published results[11] where use of French computational methods and JEF cross-section data to analyze assay data for PWR fuel with 4.5 wt % initial enrichment indicate a calculated-to-measured ratio comparable to that of lower enriched fuel.

The methodology used to combine the biases and uncertainties for individual isotopes can have a significant impact on the final k_{eff} value and needs to be properly explained and justified. Reference 9 contains a description of various recommended approaches (see Section 3 of Ref. 9) that can be used to obtain estimates of the bias and uncertainty in the SNF compositions. The simplest approach is to individually adjust the concentration of each nuclide based on the results of the validation against radiochemical assay data. This adjusted set of nuclides can then be used in the analysis of k_{eff} needed for the Safety Analysis Report. However, this process is conservative because each adjustment must be made so as to always create a more reactive system (e.g., fissile nuclides only adjusted to increase concentration and parasitic absorber nuclides only adjusted to decrease concentration).

A more realistic, but more complex approach to incorporating bias and uncertainty from the SNF compositions is to use methods[9] that demonstrate how the uncertainty in the *combined* nuclide inventory propagates to an uncertainty in the k_{eff} value. The simplest way to implement this approach would be to first obtain the set of Δk values associated with separately changing each SNF nuclide (only those used in the k_{eff} analysis) concentration by the value of the bias and uncertainty in the prediction. Reference 9 indicates that a root-mean-square (RMS) summation of these individual Δk values provides an estimate of the uncertainty in the k_{eff} value due to the combined uncertainties in the inventory prediction. The impact on k_{eff} of the bias and uncertainty from the SNF concentrations is system-dependent; thus if a fixed Δk value (RMS-combined value of Δk for all nuclides) is used to account for the nuclide inventory uncertainties, the value must be obtained based on the cask design and contents specified. Propagation of the calculated inventory uncertainties into the criticality calculation representative of the cask configurations used in the Safety Analysis Report is the reason this approach is more complex and time-consuming to implement and review.

The RMS approach assumes the uncertainty for each nuclide is independent (i.e., random) and does not consider potential correlated uncertainties in transmutation and decay chains. However, the work of Ref. 9 shows that the use of several independent "best estimate" approaches to predicting the uncertainty (e.g., use of RMS, use of Monte Carlo sampling from inventory calculated-to-measurement distributions, and direct use of measured and predicted assay data) provide similar estimates of the bias and uncertainty. This consistent estimation of the bias and uncertainty using various realistic approaches provides risk-informed confidence that the correlated uncertainties in the transmutation and decay chains have a minor impact.

Prediction of k_{eff}. Since there are not any benchmark critical experiments with commercial SNF in a cask-like environment, no one set of critical experiments provides adequate validation for burnup credit by itself. Unirradiated critical experiments have traditionally been the major source of information for criticality safety code validation and remain an excellent validation source because many have cask-like geometries and neutron-absorbing interstitial materials that simulate cask conditions. The work of Ref. 8 has used the sensitivity/uncertainty approach of Ref. 12 to demonstrate that an appropriate selection of unirradiated critical experiments can be used to validate actinide-only burnup credit applications relative to all the nuclides in Table 1 with the possible exception of ^{241}Am, which may need additional types of experiments to adequately validate.

Commercial reactor critical configurations, reactivity worth measurements, and subcritical experiments are new sources of information that have been explored as a supplement to more direct burnup credit validation data. Each of these type of experiments may be able to add to the demonstration of adequate validation for some material or geometrical aspect of a SNF cask designed for burnup credit. Reference 13 provides a discussion of the issues related to the various types of experiments as well as potential sources of proprietary and non-proprietary measurement data that may be of benefit to burnup credit. These experiments have been assessed in Ref. 8 relative to their applicability to burnup credit applications. The applicant is responsible for demonstrating that the experiments selected for the validation process are representative of the system (cask) of interest and that the code-to-experiment comparative information is utilized to estimate bounding values for the bias and uncertainty.

Integral Validation. Integral validation involves the use of depletion methods coupled with criticality calculations to determine k_{eff} for a measured system containing SNF (e.g., a spent fuel critical or reactor critical configurations). With integral validation 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 in the entire procedure are obtained. Integral validation allows for compensating errors in the depletion approach (i.e., under prediction of a given nuclide's concentration coupled with simultaneous over prediction of a different nuclide's inventory). Thus, it is desirable to ensure the uncertainty estimated for individual nuclides is understood and properly considered in the safety analysis. This situation might be of minimal concern if the experiment system is appropriately similar to the system of interest (cask). Such justification needs to be provided.

5 LICENSING-BASIS MODEL ASSUMPTIONS (RECOMMENDATION 3 OF ISG8R2)

The actinide compositions used to determine a value of k-effective for the licensing safety basis (as described in Recommendation 1) should be calculated using fuel design and in-reactor operating parameter values that appropriately encompass the range of design and operating conditions for the proposed contents. The calculation of the k-effective value should be performed using cask models, appropriate analysis assumptions, and code inputs that allow adequate representation of the physics. Of particular concern should be:

- *a) the need to account for and effectively model the axial and horizontal variation of the burnup within a spent fuel assembly (e.g., the selection of the axial burnup profiles, number of axial material zones, etc.), and*
- *b) the need to consider the potential for increased reactivity due to the presence of burnable absorbers or control rods (fully or partially inserted) during irradiation.*

The axial burnup profile database of Reference 2 provides a source of realistic, representative data that can be used for establishing a profile to use in the licensing basis safety analysis. However, care should be taken to select a profile that will encompass the range of potential k-effective values for the proposed contents, particularly near the upper end of the ranges in Recommendation 1.

A licensing basis modeling assumption where the assemblies are exposed during irradiation to the maximum (neutron absorber) loading of burnable poison rods for the maximum burnup is an appropriate analysis assumption that encompasses all assemblies that may or may not have been exposed to burnable absorbers.[3,4] 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.[5] 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 burnable poison rod exposure is less than the maximum for which burnup credit is requested, then a justification commensurate with the selected value should be provided (e.g., the lower the value, the greater the need to support the assumption with available data and/or indicate how administrative controls will prevent a misload of an assembly exposed beyond the assumed value).

Reactor Operating History and Parameter Values. The impact 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 reviewed in Section 4.2 of Ref. 13.

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. Figures 8–10 provide examples of the Δk impact seen from changes in fuel temperature, moderator temperature, and soluble boron in cask-like systems (modeled as infinite array of storage cells, but results confirmed for finite, reflected systems). All of these increases are due to the parameter increase causing a hardening of the spectrum during irradiation, thus leading to increased production of ^{239}Pu.

The impact of specific power and operating history is much more complex but has a very small impact on the cask k_{eff} value . A higher specific power provides a slightly higher k_{eff} for actinide-only burnup credit (see Fig. 11), but this trend is reversed if credit for fission products is allowed (e.g., Section 3.4.2.3 of Ref. 6 and Ref. 18 for discussion). Although the specific power at the end of irradiation is the most important, constant full-power operating histories at the desired specific power are 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 phenomena that should be assumed in the depletion model is provided in Refs. 6 and 13. Each of the trends and impacts have 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 may 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.

Also of importance is the fact that fuel demonstrated to have the highest reactivity in the unirradiated state will not necessarily be the fuel that has the highest reactivity after discharge from the reactor. 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.

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 very 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.[19] Typically, normal spatial shuffling of PWR assemblies during their life in the core would mitigate this single-cycle horizontal variation in burnup. However, if assemblies with horizontal burnup gradients observed in the database of Ref. 19 are positioned in a small cask such that the lowest burnup regions are adjacent, then increases in k_{eff} (< 1%) may be observed. Thus, the safety evaluation should address the impact of horizontal burnup gradients such as found in Ref. 19 on their cask design or demonstrate that the assemblies to be loaded in the cask will be verified to not have such gradients.

Axial Burnup Profiles. Considerable attention should be paid to the axial burnup profile(s) selected for use in the safety evaluation. Figure 12 indicates that in comparison to a uniform axial burnup assumption, a realistic axial profile gives higher cask k_{eff} values for average assembly burnups greater than about 20-25 GWd/MTU. The positive Δk from the use of an axial profile increases with burnup because the difference between assembly-averaged burnup and the burnup in the end region of the fuel increases with assembly burnup, causing the relative worth of the fuel at the ends to increase. As indicated by Fig. 12, a uniform axial profile has been found to be bounding at low burnups.

A review and evaluation of the existing, publicly available U.S. database[14] of axial burnup profiles is provided in Ref. 20. The public database[14] consists of 3169 axial burnup profiles from ~1700 different assemblies based on information from 20 different U.S. PWRs representing 106 cycles of operation through the mid-1990s. The profiles in the database include fuel designs that used burnable absorbers with different poison absorber types such as: burnable poison rods of borosilicate glass and B_4C; and integral burnable absorbers of ZrB_2, B_4C, erbium and gadolinium. In addition, the database includes assemblies exposed to control rods, including axial power shaping rods. Although the database represents only 4% of the assemblies discharged through 1994, the review indicates the database provides a good representation of discharged assemblies in terms 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, and use of burnable absorbers. The primary deficiency in the database of Ref. 14 is the number of profiles associated with assembly burnup values greater than 40 GWd/MTU and initial enrichment values greater than 4.0 wt %. However, Section 4.3 of Ref. 20 indicates that there is a high probability that profiles providing the highest reactivity in intermediate burnup ranges will also provide the highest reactivity at higher burnups. Consequently, using risk-informed judgement along with the margin presented by isotopes not included in the analysis, the existing database should be adequate for burnups beyond 40 GWd/MTU and initial enrichments above 4% if appropriate care is taken to select profiles that include a margin for the potential added uncertainty in moving to higher burnups and initial enrichments.

Previous work[21] identified the axial profiles within the database that provide the highest end effect value for each of twelve burnup groups (e.g., 38–42 GWd/MTU). This information was used to propose simulated bounding profiles for the burnup range of each group. Section 4.2.1 of Ref. 20 reports analyses that confirmed the limiting axial profiles of Ref. 21 and Fig. 13 shows the spread of k_{eff} values as reported for one burnup group. Each k_{eff} value corresponds to using a separate axial profile within the burnup group. A simulated profile proposed by Ref. 21 as an adequate bounding profile is also shown in Fig. 13 together with the mean k_{eff} value and indicators for 1, 2, and 3 standard deviations. The review in Ref. 20 revealed that, for each of the 12 burnup groups, the k_{eff} value associated with the bounding axial profile is more than 3 standard deviations above the mean and in many cases is more than 5 standard deviations above the mean. Thus, the limiting profiles are statistical outliers of those profiles in the database. However, given the finite nature of the available database (4% of the inventory through 1994 discharge), there is judged to be some low probability that some discharged SNF would have a higher reactivity than the limiting profiles identified for the same burnup group. Using a generic burnup credit cask model, a study (see Section 4.4 of Ref. 20) to investigate the impact of loading assemblies with a significantly more reactive profile (cask system worth up to 5% Δk more than the system with a limiting profile) indicated the multiplication factor for a representative burnup credit cask would increase less than ~0.5% Δk for each significantly-more-reactive assembly that is loaded in place of a limiting-profile assembly from the database. Thus, the characteristic of the limiting profiles from the database as being statistical outliers, the use of a limiting profile for all assemblies loaded in the cask, and the low consequence associated with the loading of an assembly with a higher reactivity (beyond the selected limiting profile for that burnup group) has led to the recommendation that this publicly available database is an appropriate source for selecting axial burnup profiles that will encompass the SNF anticipated for loading in a burnup credit cask.

Other sources of axial burnup profiles may be appropriate to replace or supplement the database of Ref. 14. The reviewer should assure that a description and evaluation of the database similar to that demonstrated in Refs. 20–21 has been performed. Of prime importance, 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 appropriately 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 reduced if a measurement of the axial burnup profile is performed prior to or during the cask loading operation. The measurement would need to demonstrate that the actual assembly profile is equally or less reactive than that assumed in the safety evaluation.

The ISG8R2 indicates any analysis should provide "an adequate representation of the physics." Thus, the applicant should carefully explain and justify the use of a uniform axial profile assumption in the analyses together with any Δk allowance used to accommodate the effect of the axial burnup variation. The applicant should demonstrate that the Δk value(s) properly 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. A consideration of the range of profiles anticipated for the fuel to be loaded in the cask will still be needed.

Burnable Absorbers. Assemblies exposed to fixed neutron absorbers [integral burnable absorbers (IBAs)] and removable neutron absorbers [burnable poison rods (BPRs) can have higher k_{eff} values than assemblies which are not so exposed because the presence of the absorber will harden the spectrum and lead to increased ^{239}Pu 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.

Investigations[15-16] have been performed to quantify how the k_{eff} value of a discharged assembly would change due to irradiation with BPRs and IBAs included in the assembly. A comprehensive range of assembly designs, absorber loadings, and exposure history was used to determine the impact on the k_{eff} value of SNF. The studies show that exposure to BPRs can cause the k_{eff} to increase up to 3% when the maximum number of BPRs and/or the maximum absorber loading is assumed for the maximum exposure time. More typical absorber loadings and exposures (1-cycle of 20 GWd/MTU) lead to increases of <1% Δk (e.g., see Fig. 14). By comparison, except for one IBA type where the increase was as much as 0.5% Δk (i.e., see Fig. 15), the IBAs actually provide a decrease in k_{eff} relative to assemblies not irradiated with IBAs. References 15–16 provide a base characterization for the effect of burnable absorbers on spent fuel and 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 burnable absorbers.

Control Rods. As with BPRs, control rods (CR) fully or partially inserted during reactor operation can harden the spectrum in the vicinity of the insertion and lead to increased production of ^{239}Pu. In addition, control rods can alter the axial burnup profile. In either case the control rod would have to be inserted for a reasonable 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 control rods 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 control rods in a more extensive fashion such that the impact of CR insertion would be significant.

The results of a parametric study[17] to quantify the effect of CR exposure are summarized in Fig. 16, where it can be seen that, even for significant burnup exposures (up to 45 GWd/MTU), minor axial CR insertions (e.g., < 20 cm) result in an insignificant effect (less than 0.2% Δk) on the k_{eff} value of a burnup credit cask. Control rods, if inserted, are normally placed in first cycle assemblies. However, Ref. 17 shows that full insertion for burnups up to 5–10 GWd/MTU provided an increase in cask k_{eff} values on the same order as seen for BPRs. Thus, since BPRs and CRs can not be inserted in an assembly at the same time, it follows that the inclusion BPRs in the assembly irradiation model (up to burnup values that encompasses realistic operating conditions) should adequately account for the potential increase in k_{eff} that may occur for SNF exposed to CRs during irradiation.

Control rod insertion (or use of axial power shaping rods, APSRs) during reactor operation can also lead to a distorted, or non-typical axial burnup profile. However, as noted above in the discussion of axial profiles, the existing database of axial burnup profiles[14] includes a representative sampling of assemblies exposed to CRs and APSRs. In fact, many of the limiting profiles that exist in the database are from assemblies exposed to CRs and APSRs. Thus, the appropriate selection of a limiting axial profile(s) from the available database (or one similar) would, in a risk-informed fashion, adequately encompass the potential impact for axial profile distortion caused by CRs and APSRs.

Depletion Analysis Computational Model. A review of the chart of the nuclides provides a preview of the vast number of nuclides (around a 1000) that should be tracked in the depletion and decay process to obtain an accurate estimate of the SNF concentration. Although certain nuclides that are typically tracked may not directly impact the depletion or production of the nuclides in Table 1, they can indirectly impact the production via the impact on the neutron spectrum. Tracking of a sufficiently large inventory of nuclides, the use of accurate nuclear data, and the prediction of burnup-dependent cross sections representative of the spatial region of interest are the keys to an accurate depletion analysis model. To date, most burnup credit investigations have sought to obtain spent fuel nuclide concentrations averaged horizontally over the assembly. Thus, based on comparison with assay measurements, one-dimensional physics models of PWR assembly designs have proven adequate[22,23] to predict the neutron flux spectrum at various intervals during irradiation and subsequently update the cross sections. In addition, these codes have performed well in comparisons[15,24,25] with depletion methods that use two-dimensional physics models, demonstrating that detailed geometrical modeling of the assembly and/or pin-by-pin depletion does not appear necessary for adequate prediction of k_{eff} in a cask loaded with PWR spent fuel. Such conclusions are substantiated by the fact that assembly-averaged isotopic concentrations have been used to predict reactor core critical configurations and obtained reasonable predictions (< 1% Δk) of the critical state-point.[26] Regardless of the rigor of the physics model used in depletion, it is essential that the cross sections be updated as a function of burnup (at least every 5 GWd/MTU seems adequate for calculation of the k_{eff} for SNF) and that the physics model used to update the cross sections be one that is representative of the assembly design and reactor operating history.

Models for Prediction of $k_{eff.}$ The expectations regarding the codes to be used to determine k_{eff} of a dry storage or transport cask are documented in Refs.1–3 and 27. Monte Carlo codes capable of three-dimensional (3-D) solutions of the neutron transport equation are typically required for such applications. A uniform (all assemblies assumed to have same basic characteristics) loading of SNF at a specified assembly-average burnup and initial enrichment 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 assumption that needs careful consideration. In particular, the burnup will vary rapidly at the ends of the fuel regions. Thus, the Monte Carlo 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 18–20 uniform axial regions, thus using smaller burnup zones will require some means to subdivide the burnup among the sub-zones. Studies (see Section 5.2.1 of Ref. 6 and Appendix A of Ref. 20) have 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 Ref. 20 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 an appropriate, justified 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. However, Appendix F of Ref. 6 has demonstrated that initial uniform sampling of the fuel region coupled with adequate specification of the Monte Carlo simulation (1000 particles per generation, 1000–2000 total generations) can provide properly converged results for the k_{eff} value of a SNF cask. Special strategies that may be used in the calculations to accelerate the source convergence should be carefully justified and demonstrated to be effective.

A seemingly straightforward, but important issue is the need to verify that the correct SNF composition associated with the depletion/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 (in the correct units) from the depletion/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. Verification of the interface process and/or the computer code used to automate the data handling should be performed.

6 LOADING CURVE (RECOMMENDATION 4 OF ISG8R2)

Loading Curve. A loading curve is a plot that demonstrates, as a function of initial enrichment, the assigned burnup value above which fuel assemblies may be loaded in the cask. 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 cask loading. The applicability of the loading curve to bound various fuel types or burnable absorber loadings should be justified. To limit the opportunity for misloading, only one loading curve should be used for each cask loading.

Typically the personnel responsible for loading a SNF cask have ready knowledge of the average assembly burnup and initial enrichment values. Thus, a curve that provides the burnup and initial enrichment combination associated with the upper subcritical limit (or USL, see Ref. 27) for the cask will provide a rapid means to assess whether a specific assembly is acceptable for loading in the cask. Such a curve is called a loading curve and the preparation of such a curve requires numerous calculations with variable burnup/enrichment combinations to determine sufficient points for the curve. The reviewer should assure that the process used to generate the loading curve is explained thoroughly and should further verify that the loading curve is representative of or below the cask k_{eff} value associated with the USL. The STARBUCS sequence[28] of the SCALE-5 system is a computational tool that can be used to help verify the adequacy of a loading curve.

Figure 17 presents representative loading curves. The discontinuities in the loading curve represent the locations where different axial burnup profiles were used to account for profile changes with burnup. A different loading curve will occur based on the assumptions used in the analyses. Reference 29 provides additional loading curves illustrating the anticipated impact of various assumptions on the cask loading curve. Each loading curve should be clearly marked relative to key assembly characteristics (e.g., assembly design type, cooling time, etc.).

7 ASSIGNED BURNUP LOADING VALUE (RECOMMENDATION 5 OF ISG8R2)

Administrative procedures should be established to ensure that the cask will be loaded with fuel that is within the specifications of the approved contents. The administrative procedures should include a measurement that confirms the reactor record for each assembly. Procedures that confirm the reactor records using measurement of a sampling of the fuel assemblies will be considered if a database of measured data is provided to justify the adequacy of the procedure in comparison to procedures that measures each assembly.

The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For confirmation of assembly reactor burnup record(s), the measurement should provide agreement within a 95% confidence interval based on the measurement uncertainty. 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.

The loading curve must be used in conjunction with other criteria to determine if a fuel assembly may be loaded into the cask. For example, if proper consideration of BPRs is not given, the loading curve criteria should include an appropriate exclusion. Thus, it should be verified that all restrictions on cask loading, consistent with the assumptions used in the evaluation, are clearly specified in the safety evaluation report and that cask loading procedures have included checks against criteria needed to ensure only approved contents are loaded. Applicants should provide a list of criteria to be confirmed prior to or during loading. Adequate justification and demonstration of the criteria values (and the rationale for omitting certain criteria) should be provided.

A measurement that is able to confirm the average burnup recorded for an assembly is needed. The administrative procedures for cask loading should include such a measurement and note that the uncertainty in the measurement uncertainty and the uncertainty in the reactor records should both be included in adjusting the reactor record burnup to an assigned burnup loading value. The burnup measurement approaches proposed to date use measurements of numerous assemblies and comparisons against reactor record values to self-calibrate the system. Thus, the measurement and record for these types of systems are not independent and the uncertainty in both the records and the measurement should be considered in order to mitigate the potential for a systematic error in the reactor records. An assessment of the uncertainty of the burnup values provided in reactor records has been performed,[30] indicating uncertainties should be less than 5% for PWR assemblies.

In Regulatory Guide 3.71, NRC endorsed the recommendations of ANSI Standard 8.17-1997 with the exception that credit for fuel burnup may be take only when the amount of burnup is confirmed by physical measurements. Any request for a plan to measure a random sample of fuel assemblies in lieu of measuring every assembly needs to be justified by a measurement database and specific procedures for executing the plan. Requests for sampling need to consider the demonstrated accuracy of the burnup record system as confirmed in the measurement data base.

8 ESTIMATE OF ADDITIONAL REACTIVITY MARGIN (RECOMMENDATION 6 OF ISG8R2)

Estimate of Additional Reactivity Margin. The available experimental database relevant to use of burnup credit in the safety analysis of a PWR cask is not as extensive as the database available to support licensing with the unirradiated fuel assumption. The process of assuring that appropriate values and conditions have been applied in the safety analysis is also more difficult. For example, there may be uncertainties that are not directly evaluated in the modeling or validation processes for actinide-only burnup credit (e.g., k-effective validation uncertainties caused by a lack of critical experiment benchmarks with either actinide compositions that match those in spent fuel or material distributions that represent reactive ends of spent fuel in casks). Also, there may be potential uncertainties in the models that calculate the licensing-basis actinide inventories (e.g., caused by any outlier assemblies with higher-than-modeled reactivity such as may be caused by prolonged use of control rod insertion during irradiation, axial profiles not encompassed by the data of Reference 2, or exposure to unanticipated operating conditions that increase reactivity). Decisions on the adequacy of the safety analysis relevant to these difficult-to-quantify uncertainties are more straightforward if design-specific analyses are provided that estimate the additional reactivity margins available from absorber nuclides (fission product and actinides) not included in the licensing safety basis (as described in Recommendation 1). The reviewer should assess the *estimated reactivity margins to determine their adequacy for offsetting any potential uncertainties introduced by the type of effects discussed above.*

As indicated in Table 3, the assay data available for fission product nuclides is scarce relative to the data available for the actinides of Table 1. In addition, the type of experiments (critical experiments, worth experiments, etc.) that may be needed to validate the reactivity effect from fission products are generally not publically available and/or difficult to use (e.g., reactor critical measurements and worth measurements). Thus, until additional data are available to validate the quantity of the fission product worth for a specific cask, it is not recommended that the fission product inventory be considered in the licensing basis safety analysis for burnup credit.

The fact that the neutron-absorbing properties of fission products are known to reduce the k_{eff} value beyond the actinide-only assumption indicates that the actinide-only assumption is conservative. However, quantity of the conservatism can not be well substantiated given the existing experiment and measurement data. Until additional experience is gained with the uncertainties associated with actinide-only burnup credit, an estimate of the additional reactivity margin that is available from nuclides not considered in the safety analysis may be used to compensate for uncertainties not readily understood or quantified in the actual safety analysis using the actnides of Table 1. The estimate should be specific to the cask design since the margin will vary depending on the external absorbers in the cask basket. To help confirm the adequacy of the estimate, the applicant may refer to the estimates provided in Ref. 31 or Refs. 32–34. The estimation of additional reactivity margin should not be used to reduce the level of validation or realistic bounding assumptions used as a basis for safety. However, the information can be used to help justify that difficult-to-quantify uncertainties are adequately covered within the safety envelope of the cask design. Other easily identified conservative assumptions that may have been used in the licensing basis model can also be considered.

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