REVIEW OF TECHNICAL STUDIES IN THE UNITED STATES IN SUPPORT OF BURNUP CREDIT REGULATORY GUIDANCE

John C. Wagner, Cecil V. Parks, Don E. Mueller, and Ian Gauld

Oak Ridge National Laboratory[*]
P.O. Box 2008, Bldg. 5700, MS-6170
Oak Ridge, TN 37831-6170 USA
wagnerjc@ornl.gov

Taking credit for the reduction in reactivity associated with fuel depletion can enable more cost-effective, higher-density storage, transport, disposal, and reprocessing of spent nuclear fuel (SNF) while maintaining sufficient subcritical margin to establish an adequate safety basis. Consequently, there continues to be considerable interest in the United States (U.S.), as well as internationally, in the increased use of burnup credit in SNF operations, particularly related to storage, transport, and disposal of commercial SNF. This interest has motivated numerous technical studies related to the application of burnup credit, both domestically and internationally, as well as the design of SNF storage, transport and disposal systems that rely on burnup credit for maintaining subcriticality. Responding to industry requests and needs, the U.S. Nuclear Regulatory Commission (NRC) initiated a burnup credit research program in 1999, with support from the Oak Ridge National Laboratory (ORNL), to develop regulatory guidance and the supporting technical bases for allowing and expanding the use of burnup credit in pressurized-water reactor SNF storage and transport applications. Although this NRC research program has not been continuous since its inception, considerable progress has been achieved in many key areas in terms of increased understanding of relevant phenomena and issues, availability of relevant information and data, and subsequently updated regulatory guidance for expanded use of burnup credit. This paper reviews technical studies performed by ORNL for the U.S. NRC burnup credit research program. Examples of topics include reactivity effects associated with reactor operating characteristics, fuel assembly characteristics, burnable absorbers, control rods, spatial burnup distributions, cooling time, and assembly misloading; methods and data for validation of isotopic composition predictions; methods and data for validation of criticality calculations; and operational issues and data related to assembly burnup confirmation. The objective of this paper is to summarize the work and significant accomplishments, with references to the technical reports and publications for complete details, and provide a useful resource to others in the burnup credit community.

I. INTRODUCTION

Historically, criticality safety analyses for transport and dry cask storage of spent nuclear fuel (SNF) in the United States (U.S.) assumed the fuel contents to be unirradiated (i.e., "fresh" fuel) compositions. However, it is well understood that taking credit for the reduction in reactivity associated with fuel depletion can enable more cost-effective, higher-density storage, transport, disposal, and reprocessing of SNF while maintaining sufficient subcritical margin to establish an adequate safety basis. In recent decades, increasing SNF inventories (see Figure 1) has necessitated expanding and optimizing SNF storage and transport capacity. Consequently, there has been, and continues to be, considerable interest in

[*]This manuscript has been authored by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy. The United States Government retains and the publisher, by accepting the article for publication, acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes.

the United States, as well as internationally, in the increased use of burnup credit in SNF operations, particularly related to storage, transport, and disposal of commercial pressurized-water reactor (PWR) SNF. In contrast, far less interest and far fewer studies have been devoted to burnup credit for boiling-water reactor (BWR) SNF. This is largely due to a lesser need for burnup credit, as compared to PWR SNF, to enable high-density storage and transport configurations.

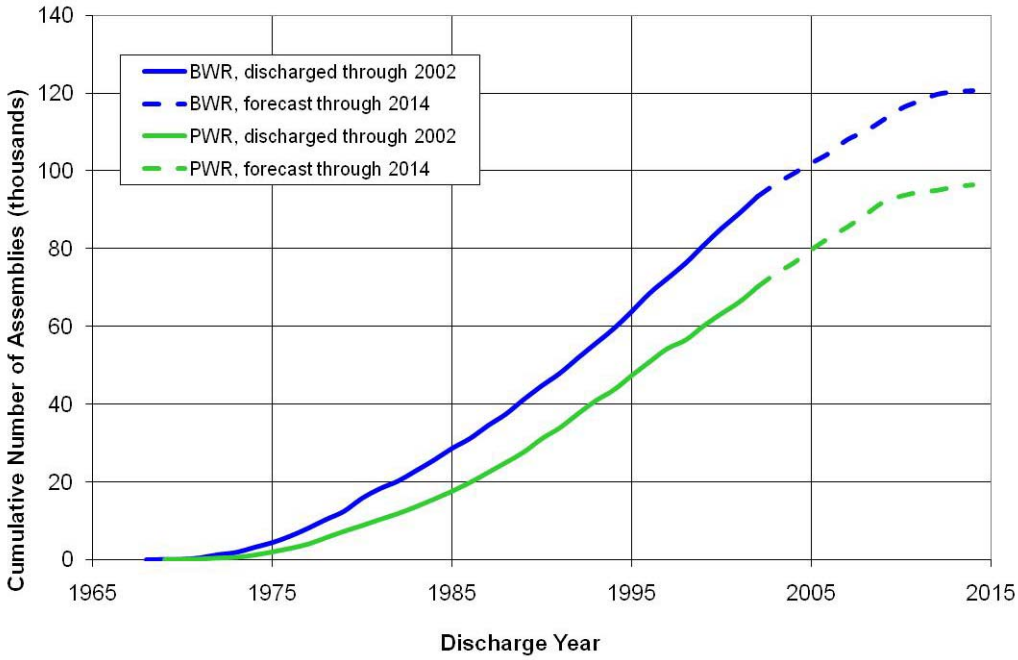

Figure 1. U.S. discharged BWR and PWR SNF assemblies (with forecast through 2014[†]). *Data Source*: Ref. 1.

In July 1999, the U.S. Nuclear Regulatory Commission (NRC) Spent Fuel Project Office (SFPO) issued Interim Staff Guidance 8, Revision 1 (ISG8R1), to provide recommendations for the use of burnup credit in storage and transport of PWR spent fuel [2]. A discussion of the technical considerations that helped form the development of ISG8R1 is available in Ref. 3. Note, ISG8R1 is specific to PWR fuel and no such similar guidance permitting burnup credit for BWR fuel in storage and transport has been developed. The ISG8R1 recommendations were subsequently included in the Standard Review Plan for transportation casks and dry storage cask facilities [4, 5]. Subsequent to the issuance of ISG8R1, the NRC Office of Nuclear Regulatory Research (RES) initiated an effort at the Oak Ridge National Laboratory (ORNL) to investigate the technical basis for extending the criteria and recommendations of ISG8R1 with the goal of improved implementation of burnup credit. The work sponsored by NRC RES provided reference material for NRC SFPO to use in their preparation of Revision 2 of ISG8 (ISG8R2) [6], which was released in September 2002. More recently, work sponsored by NRC RES has been focused on resolution of issues related to burnup credit license applications and extension of burnup credit to include credit for fission products.

[†] The discharge forecasts were estimated nearly ten years ago and may not accurately represent actual discharges, especially given the large number of plant life extensions that have been granted in recent years.

Similar to ISG8R1, the recommendations provided in ISG8R2 cover six areas:

(a) limits for the licensing basis,
(b) guidance on code validation,
(c) guidance on licensing-basis model assumptions,
(d) guidance on preparation of loading curves,
(e) the process for assigning a burnup loading value to an assembly, and
(f) estimation of additional reactivity margin beyond that which can be substantiated through the validation process.

Reference 7 discusses each of the six recommendations within ISG8R2 with specific emphasis on the changes implemented with ISG8R2 and the technical basis for the changes.

Although the NRC research program has not been continuous during the past ten-plus years, considerable progress has been achieved in many key areas in terms of increased understanding of relevant phenomena and issues, availability of relevant information and data, and subsequently updated regulatory guidance for expanded use of burnup credit. The following sections provide a brief chronological review of the technical studies performed by ORNL for the U.S. NRC burnup credit research program. Examples of topics covered include reactivity effects associated with reactor operating characteristics, fuel assembly characteristics, burnable absorbers, control rods, spatial burnup distributions, cooling time, and assembly misloading; methods and data for validation of isotopic composition predictions; methods and data for validation of criticality calculations; and operational issues and data related to assembly burnup confirmation. The objective of this paper is to summarize the work and significant findings, with references to the technical reports and publications for complete details, and provide a useful resource to others in the burnup credit community.

II. REVIEW OF TECHNICAL STUDIES

With the criteria and guidance of ISG8R1 established by the NRC licensing staff, the research program was directed to identify and perform work needed to develop expanded guidance relative to selected elements of ISG8R1, implement software enhancements that can facilitate burnup credit safety analyses, and develop the technical basis for the NRC SFPO to use in considering future revisions of ISG8. A baseline report [8] was developed in 1999 to document the current status of burnup credit and to provide a strawman prioritization for areas where additional guidance, information, and/or improved understanding were judged beneficial to the effective implementation of burnup credit in transport and dry storage casks. The prioritization considered input obtained at public workshops sponsored by the NRC and the Nuclear Energy Institute (NEI).

As a result of the initial review and input from industry and licensing staff, the focus areas for the NRC research program were established [9]:

1. Development of a comprehensive reference report that uses current cask designs (rail and truck) to provide a consistent basis for demonstrating the magnitude of the various negative reactivity components as a function of burnup, initial enrichment, and cooling time.
2. Development of an automated process for coupling the depletion/decay analysis to the criticality analysis to support initial license reviews.
3. Development of a computational benchmark for a generic rail cask design to support the calibration of an applicant's estimation of fission product margin per ISG8R1 recommendations.
4. Development of criteria and guidance for the selection of an appropriate axial profile for use in the safety assessment.

5. Development of an initial recommendation and associated technical basis for potential near-term modifications to ISG8R1 relative to the use of cooling times other than 5 years.
6. Development of an initial recommendation and associated technical basis for potential near-term modifications to ISG8R1 relative to the use of burnup credit with PWR fuel containing burnable poison rods and/or integral burnable absorbers.
7. Investigation of the potential for modifying or removing the loading offset in ISG8R1 (the added burnup margin required for fuel with initial enrichments above 4.0 wt %) based on existing and potential experimental data.
8. Review and evaluation of existing and proposed experimental data to (a) demonstrate and rank the relevance of experiments for methods validation using quantitative criteria, (b) identify experimental needs, and (c) assess technical bases for "certifying" a minimum reactivity margin associated with fission products.

The first two of these eight areas were aimed at assisting the licensing staff in preparation for review of applications that use burnup credit in the safety analysis. Areas 3 and 4 were included to provide additional guidance to assist in effective implementation of the ISG8R1 recommendations. The remaining areas were directed at expanding the inventory of fuel that would be allowed in a burnup-credit cask. Results of efforts in each of these eight areas, as well as in tangential areas, are discussed in the following sections, which are organized chronologically and correspond to published NUREG/CR reports.

II.A Review and Prioritization of Technical Issues

Initial efforts focused on reviewing the available literature to establish a baseline for the current level of knowledge and understanding and support the development of a research plan for addressing the technical issues in a prioritized manner. These efforts produced a reference report [8] that reviewed the relevant background information, previous works, and technical issues necessary to initiate and facilitate a PIRT (Phenomena Identification and Ranking Tables) process for the use of burnup credit in light-water reactor (LWR) spent fuel storage and transport cask applications. The PIRT process was used by the NRC RES to help prioritize a coordinated program of research and as a means to obtain input/feedback from industry and other interested parties. The review and discussion in the reference report were based on knowledge and experience gained from work performed in the United States and other countries. Current regulatory practice and perceived industry needs were also reviewed as a background for prioritizing technical needs that would facilitate safe practice in the use of burnup credit. Relevant physics and analysis phenomena were identified, and an assessment of their importance to burnup credit implementation was developed. Finally, phenomena that needed to be better understood for effective licensing, together with technical issues that require resolution, were presented and discussed in the form of a prioritization ranking and an initial draft program plan. The reference report was used in the NRC PIRT process, and results of the PIRT panel's findings are documented in Ref. 10.

II.B Reactivity "Equivalencing" Practice in Spent Fuel Pools

Prior to proceeding to address the aforementioned focus areas, a related activity [11] was initiated to examine the practice of equating the reactivity of spent fuel to that of fresh fuel for the purpose of performing burnup credit criticality safety analyses for PWR spent fuel pool (SFP) storage conditions. The investigation consisted of comparing effective neutron multiplication factor, k_{eff}, estimates based on reactivity "equivalent" fresh fuel enrichment (REFFE) to k_{eff} estimates using calculated spent fuel isotopic compositions. Analyses of selected storage configurations common in PWR SFPs showed that the practice yields nonconservative results (on the order of a few tenths of a percent) in configurations in which the spent fuel is adjacent to higher reactivity assemblies (e.g., fresh or lower-burned assemblies) and yields conservative results in configurations in which spent fuel is adjacent to lower-reactivity assemblies (e.g., higher-burned fuel or empty cells). When the REFFE was determined based on

unborated water moderation, analyses for storage conditions with soluble boron present revealed significant nonconservative results associated with the use of the REFFE. This observation was considered important, especially considering the allowance of credit for soluble boron up to 5% in reactivity [12]. Finally, it was shown that the practice of equating the reactivity of spent fuel to fresh fuel is acceptable, provided the conditions for which the REFFE was determined remain unchanged. Determination of the REFFE for a reference configuration and subsequent use of the REFFE for different configurations violates the basis used for the determination of the REFFE and, thus, may lead to inaccurate, and possibly nonconservative, estimates of reactivity. A significant concentration (~2000 ppm) of soluble boron is typically (but not necessarily required to be) present in U.S. PWR SFPs, of which only a portion may be credited in safety analyses. Thus, a subcritical margin exists that should more than account for errors or uncertainties associated with the use of the REFFE. Consequently, the findings did not represent a significant safety concern unless/until the subcritical margin associated with the soluble boron (that is not currently explicitly credited) is offset by the uncertainties associated with burnup credit and/or the expanded allowance of credit for the soluble boron. The full details of this study are available in Refs. 11 and 13.

II.C Computational Benchmark

Returning to the research plan, the next effort was to develop and document a computational benchmark [14] for the estimation of the additional reactivity margin available in SNF from fission products and minor actinides in a burnup credit storage/transport environment, relative to SNF compositions containing only the major actinides. The benchmark problem/configuration was developed to be representative of a large, rail-type burnup credit cask designed to hold 32 PWR assemblies and is referred to as the GBC-32 cask. The purpose of the computational benchmark was to provide a reference configuration for the estimation of the additional reactivity margin, which is recommended in ISG8R1, and document reference estimations of the additional reactivity margin as a function of initial enrichment (2–5 wt % ^{235}U), burnup (0–60 GWd/MTU), and cooling time (0–40 years). Consequently, the geometry and material specifications were provided in sufficient detail to enable independent evaluations. Estimates of additional reactivity margin for this reference configuration may be compared with those of similar burnup-credit casks to provide an indication of the validity of design-specific estimates of fission-product margin. For example, the minimum additional reactivity margin available from fission products and minor actinides was quantified for the burnup, initial enrichments, and cooling times considered, and the minimum values were shown (see Figure 2) to occur at zero cooling time and increase as a function of burnup from ~0.03 Δk at 10 GWd/MTU to ~0.08 Δk at 60 GWd/MTU. The reference solutions were generated with the SAS2H-depletion and CSAS25-criticality sequences of the SCALE 4.4a package. Although the SAS2H and CSAS25 sequences have been extensively validated in other works, the reference solutions were not directly or indirectly based on experimental results. Consequently, the computational benchmark cannot be used to satisfy the ANS 8.1 requirements for validation of calculational methods and was not intended to be used to establish biases for burnup credit analyses.

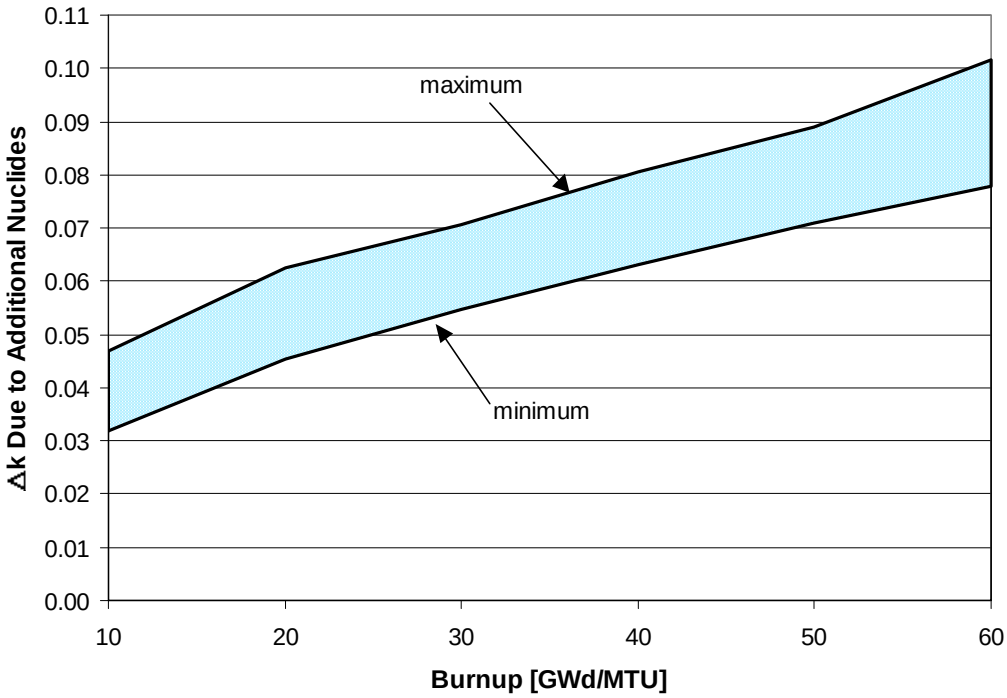

Figure 2. Range of Δk values in the GBC-32 cask due to the additional nuclides (minor actinides and fission products) as a function of burnup for all cooling times and initial enrichments considered. *Source*: Ref. 14.

II.D Automated Burnup Credit Computational Sequence

To facilitate analyses to investigate and quantify the effects of the various burnup credit input assumptions, as well as to support independent calculations for license reviews, a new computational analysis sequence, STARBUCS (Standardized Analysis of Reactivity for Burnup Credit using SCALE), was developed [15] for performing automated burnup credit criticality safety analyses. The sequence automates a depletion calculation for each of the burnup-dependent regions of a spent fuel assembly, or other system containing depleted fuel, using the ORIGEN-ARP sequence of SCALE [16]. The spent fuel compositions are then used to generate resonance self-shielded cross sections for each region of the problem, which are applied in a three-dimensional Monte Carlo criticality safety calculation using either the KENO V.a or KENO-VI code. This burnup credit analysis sequence allows the user to easily model the axial and horizontal burnup gradients in a spent fuel assembly, select the specific actinides and/or fission products to be included in the criticality analysis, and apply isotopic correction factors to the predicted spent fuel nuclide inventory to account for calculational bias and uncertainties. Although STARBUCS was developed to address the burnup credit analysis needs for spent fuel transport and storage applications, it provides sufficient flexibility to allow criticality safety assessments for other configurations involving SNF. STARBUCS was released to the public with the SCALE 5.1 code package and has since been enhanced to include a capability to automatically generate loading curves and the ability to perform continuous energy Monte Carlo criticality calculations [16,17].

II.E Burnable Neutron Absorbers

The presence of fixed burnable neutron absorbers [integral burnable absorbers (IBAs)] and removable neutron absorbers [burnable poison rods (BPRs)] in an LWR fuel assembly lattice during depletion hardens the neutron spectrum, resulting in increased production of fissile plutonium isotopes and reduced

^{235}U depletion. In addition, when removable neutron absorbers are inserted, the spectrum is further hardened due to displacement of the moderator. Consequently, a fuel assembly burned in conjunction with burnable absorbers may have a higher reactivity for a given burnup than an assembly that has not used burnable absorbers. Therefore, where fixed absorbers have been employed, computational predictions of SNF reactivity must properly consider the impact of their presence.

Due to the lack of quantitative information on the effect of removable neutron absorbers, ISG8R1 recommended restricting the use of burnup credit to assemblies that had not used burnable absorbers. This recommended restriction eliminated a large portion of the discharged spent fuel assemblies from cask loading, and thus severely limited the practical usefulness of burnup credit. Hence, under the NRC RES research program, investigations [18, 19, 20] were 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. Details on several relevant BPR and IBA designs used in the United States are included in Refs. 18 and 19, respectively. The studies showed that exposure to BPRs can cause the k_{eff} to increase by up to 3% when the maximum absorber loading is assumed for the maximum exposure time. More typical absorber loadings and exposures (one cycle of 20 GWd/MTU) lead to increases of <1% Δk (e.g., see Figure 3). By comparison, except for one IBA type [integral fuel burnable absorber (IFBA)], where the increase was as much as 0.5% Δk, the IBAs actually provide a decrease in k_{eff} relative to assemblies not irradiated with IBAs. References 18, 19, and 20 provide a characterization of 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. This result led to the recommendation included in ISG8R2 allowing assemblies exposed to burnable absorbers to be loaded in a burnup-credit cask.

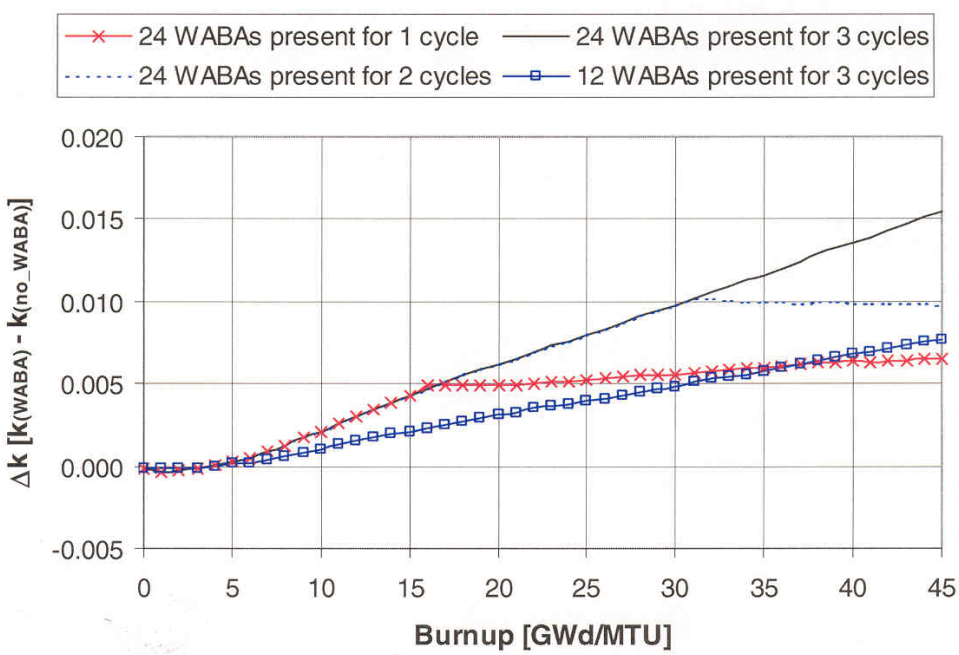

Figure 3. Comparison of Δk values, as a function of burnup, for assemblies exposed to wet annular burnable assembly (WABA) rods. Results correspond to Westinghouse 17 × 17 assemblies with 4.0 wt % ^{235}U initial enrichment. *Source*: Ref. 18.

II.F Control Rods

As with BPRs, control rods (CRs) 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 and reduced ^{235}U depletion. In addition, CRs can alter the axial-burnup profile. In either case, the CR would have to be inserted for a reasonable fraction of the total irradiation time for these effects to be seen in terms of an increase in SNF reactivity. Domestic PWRs typically do not operate with CRs inserted, although the tips of the rods may reside near the fuel ends. However, some older domestic reactors and certain foreign reactors may have used control rods more extensively such that the impact of CR insertion would be significant.

In the absence of readily available information on the extent of CR usage in U.S. PWRs and the subsequent reactivity effect of CR exposure on discharged SNF, ISG8R1 recommended restricting the use of burnup credit to assemblies that had not used CRs. To address this issue, a parametric study of the effect of CR exposure on the reactivity of discharged SNF for various CR designs, including Axial Power Shaping Rods (APSRs), fuel enrichments, and exposure conditions (i.e., burnup and axial insertion) was performed [20, 21]. The study was performed in two parts. In the first part, two-dimensional assembly calculations were performed, effectively assuming full axial CR insertion. The calculations bound the effect of CR exposure and facilitate comparisons of the various CR designs. In the second part, three-dimensional calculations were performed to quantify the reactivity effect of CR exposure in a burnup credit cask environment and determine the effect of partly inserted CRs. The reactivity effect as a function of axial insertion depth is shown in Figure 4, 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. The results from the study demonstrate that the reactivity effect increases with increasing CR exposure (e.g., burnup) and decreasing initial fuel enrichment (for a fixed burnup). Further, CR exposure was shown to have a larger effect on discharge reactivity when it occurs later in the assembly burnup. For variations in CR design, there exists a direct relationship between the reactivity worth of the CRs and their effect on discharge reactivity—higher reactivity worth CRs result in larger effects on discharge reactivity. The effects were quantified and typical operating conditions were reviewed, enabling an increased understanding of the effect of CR exposure on the reactivity of discharged SNF. The study showed that full CR insertion for burnup values up to 5–10 GWd/MTU results in an increase in cask k_{eff} values on the same order as seen for BPRs. Thus, since BPRs and CRs cannot be inserted in an assembly at the same time, it follows that the inclusion of BPRs in the assembly irradiation model (up to burnup values that encompass realistic operating conditions) should adequately account for the potential increase in k_{eff} that may occur for SNF exposed to CRs during irradiation. This result led to the recommendation included in ISG8R2 allowing assemblies exposed to CRs to be loaded in a burnup-credit cask.

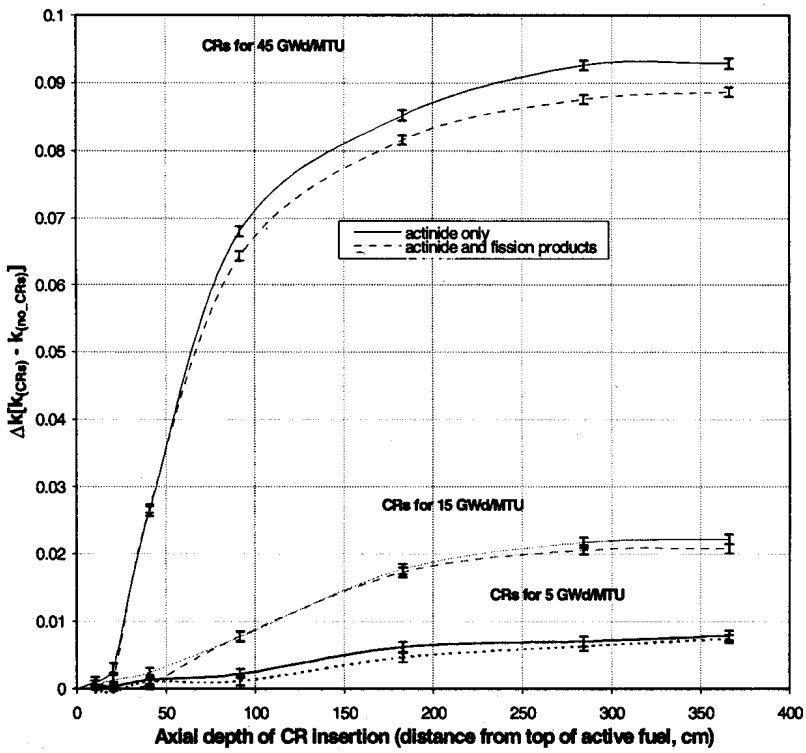

Figure 4. Impact of CR insertion during irradiation on SNF in the GBC-32 cask. *Source*: Ref. 21.

II.G Cooling Time

ISG8R1 recommended limiting licensing-basis burnup credit analyses to an out-of-reactor cooling time of five years. This recommendation eliminates assemblies with shorter cooling times from cask loading and limits the allowable credit for reactivity reduction associated with cooling time. To evaluate the potential for expanding this recommendation, a comprehensive study [22] of the reactivity behavior as a function of cooling time for various cask designs and SNF compositions was performed. The benefits of additional credit for cooling time were quantified based on a realistic high-capacity rail-type cask designed for burnup credit, i.e., the GBC-32 cask. While the report was primarily focused on cask storage and transportation, analyses were extended out to 100,000 years to understand the relevant concerns associated with long-term disposal and their possible influence on storage and transportation practice. Figure 5 illustrates the expected reactivity behavior for SNF in the GBC-32 cask for different SNF composition assumptions, including actinides only, actinides plus fission products, and all available nuclides. The fact that the reactivity begins to increase around 100 years after discharge necessitates consideration of the time frame for interim SNF storage and transport in the evaluation of acceptable cooling times. The curve indicates that the reactivity of the SNF at 40 years after discharge is approximately the same as that of the SNF cooled for 200 years. At the time of this study, the probability that SNF in a storage or transportation cask would remain in place for more than 200 years was judged to be low, which led to the recommended limiting cooling-time criterion in ISG8R2 of 40 years (i.e., no credit for cooling time beyond 40 years should be taken). Given the recent U.S. decision to withdraw the license application for the proposed repository at Yucca Mountain, the probability that SNF will remain in a storage or transportation cask for greater than 200 years has increased, and hence the current ISG8R2 recommendation allowing out-of-reactor cooling for a time period between 1 and 40 years may need to be reassessed.

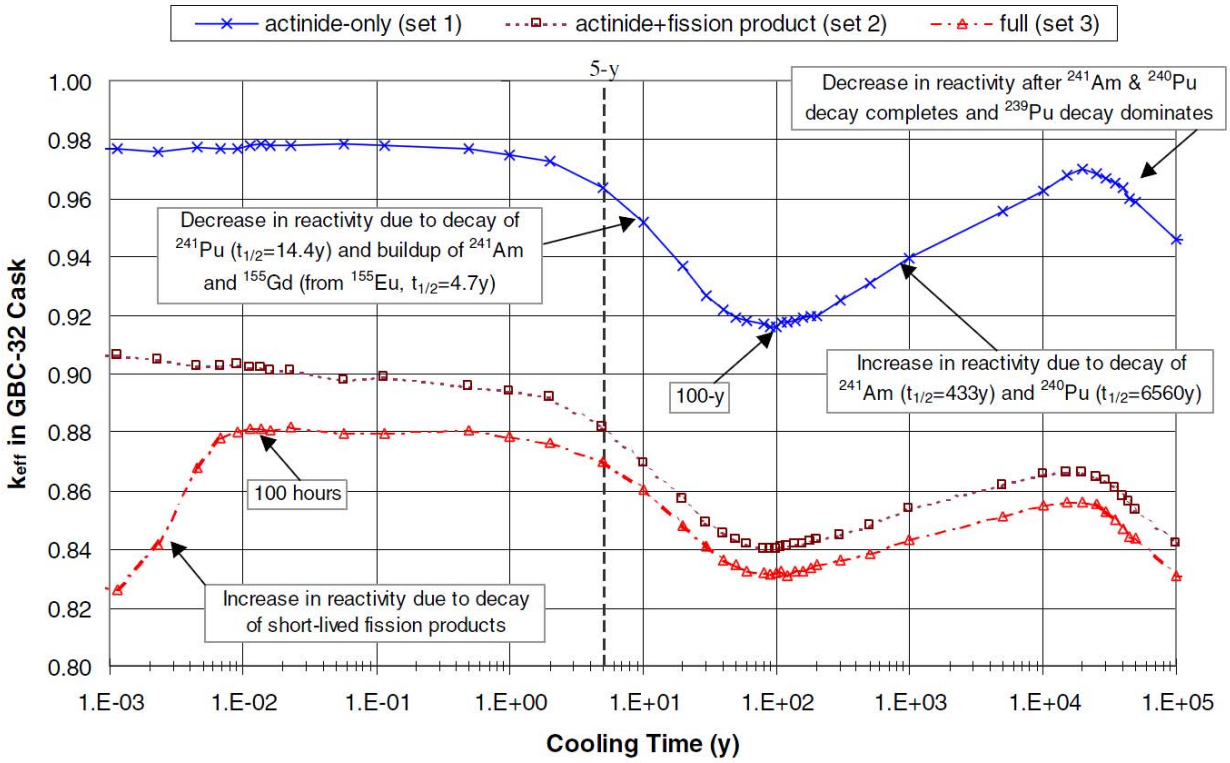

Figure 5. Reactivity behavior in the GBC-32 cask as a function of cooling time for different nuclide sets. The calculations correspond to fuel with 4.0 wt % ^{235}U initial enrichment that has accumulated 40 GWd/MTU burnup. *Source*: Ref. 22.

II.H Axial Burnup

The axial-burnup profile has a significant impact on reactivity (see Figure 6) [23] and therefore is an important component of a burnup-credit safety analysis. To support improved guidance in ISG8R2, a review and evaluation of the publicly available U.S. database [24] of axial-burnup profiles were performed [25]. The effect of the axial-burnup distribution on reactivity and proposed approaches for addressing the axial-burnup distribution were also reviewed and documented. The U.S. database of profiles was examined in detail to identify profiles that maximize the k_{eff}, assess its adequacy for PWR burnup credit analyses, and investigate the existence of trends with fuel type and/or reactor operations. Although the U.S. database represents only 4% of the assemblies discharged through 1994, the review indicated that 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. 24 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, Ref. 25 indicates that a high probability exists that profiles providing the highest reactivity in intermediate burnup ranges will also provide the highest reactivity at higher burnups. Consequently, by using risk-informed judgment along with the margin presented by isotopes not included in the analysis, the existing database was judged 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.

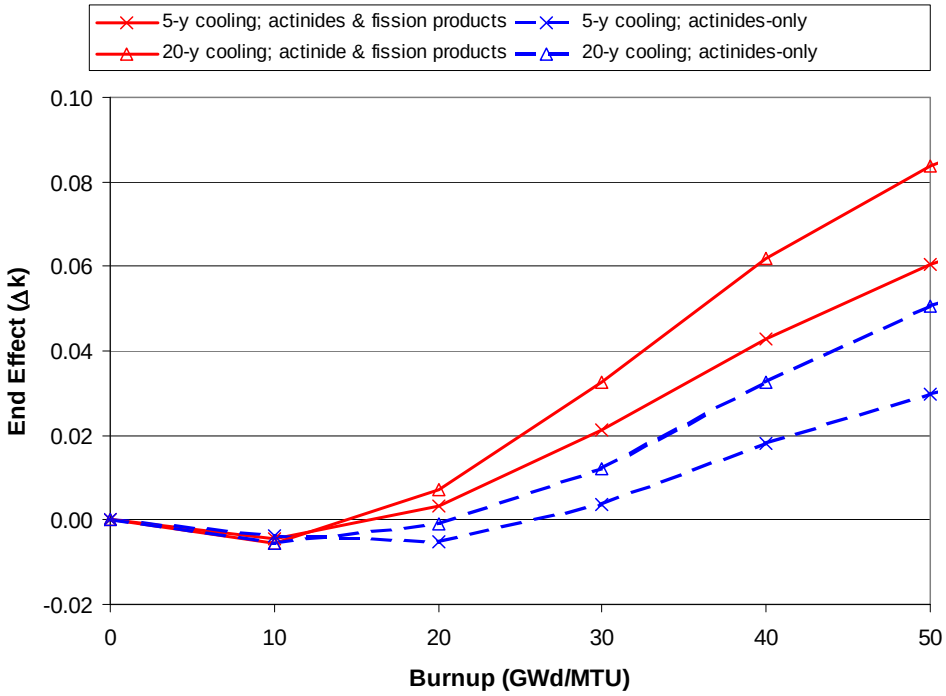

Figure 6. Example of end effect in the GBC-32 cask as a function of burnup for two cooling times with and without fission products present. The calculations correspond to fuel with 4.0 wt % ^{235}U initial enrichment. *Source*: Ref. 25.

However, given the finite nature of the available database (4% of the inventory through 1994 discharge), a low probability exists that some discharged SNF would have a higher reactivity than the limiting profiles identified for the same burnup group. Using the GBC-32 cask model, Ref. 25 investigated the impact of loading single assemblies with a significantly more reactive profile and found the consequence on reactivity to be small. Thus, the characterization of the limiting profiles from the database as statistical outliers, the use of a limiting profile for all assemblies loaded in a 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) have led to the recommendation that this publicly available database be accepted as an appropriate source for selecting axial-burnup profiles that will encompass the SNF anticipated for loading in a burnup-credit cask.

Note that the insertion of CRs (or use of APSRs) during reactor operation can lead to a distorted, or nontypical, axial-burnup profile. As noted above, the existing database of axial-burnup profiles [24] 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 [25]. Thus, the appropriate selection of a limiting axial profile(s) from the available database (or a similar one) was judged to adequately encompass the potential impact for axial-profile distortion due to CRs and APSRs.

II.I Impacts of Analysis Assumptions on Loading Curves

Having addressed the specific priority issues and limitations in ISG8R1, efforts focused on studies to assess reactivity margins and loading curves for PWR burnup-credit criticality safety evaluations. Whereas previous studies quantified the impact of various calculational assumptions on predicted k_{eff} values for various SNF systems, these studies systematically varied the individual calculational (depletion

and criticality) assumptions to demonstrate the impact on the predicted k_{eff} values and burnup-credit loading curves for the GBC-32 cask. The purpose of the studies was to provide a greater understanding of the importance of input parameter variations and quantify the impact of calculational assumptions on the outcome of a burnup-credit evaluation (i.e., loading curves). The report [26] documenting these studies provided guidance to regulators and industry on the technical areas where improved information could most enhance the estimation of accurate subcritical margins and identified areas where future work would provide the most benefit. The report also included an evaluation of the degree of burnup credit needed for high-density casks to transport the current discharged SNF inventory. By comparing PWR discharge data to actinide-only-based loading curves and determining the number of assemblies that meet the loading criteria, the evaluation showed that additional negative reactivity (through either increased credit for fuel burnup or cask design/utilization modifications) was necessary to accommodate the majority of the discharged SNF assemblies in high-capacity casks. The calculated actinide-only-based loading curves are such that a notable portion of the SNF inventory would be unacceptable for loading because the burnup value is too low for the initial enrichment (see Table 1). It was also shown that relatively small shifts in an actinide-only-based cask loading curve, which increase or decrease the minimum required burnup for a given enrichment, can have a significant impact on the number of SNF assemblies that are acceptable for loading (see Figure 7). Thus, as the uncertainties and corresponding conservatisms in burnup credit analyses are better understood and reduced, the population of SNF acceptable for loading in high-capacity casks will increase. Based on these results, it was suggested that future work focus on improving the accuracy associated with estimates of the subcritical margin with burnup credit with the goal of substantially increasing the number of assemblies that are acceptable for loading in high-capacity casks. For example, alternative approaches for validation of the isotopic compositions should be considered. The most significant component that would improve accuracy, and subsequently enhance the utilization of burnup credit, is the accurate inclusion of fission products. Consequently, experimental data and an effective approach for validation of fission products were confirmed to be key elements necessary for the expansion of burnup credit.

Table 1. Summary of SNF acceptability in the GBC-32 cask with actinide-only burnup credit for the four assembly types considered (*Source*: Ref. 27)

Assembly type	Total in discharge data	Number acceptable for loading	Number unacceptable for loading
CE 14×14	5,453	4,194 (77%)	1,259 (23%)
B&W 15×15	6,439	190 (3%)	6,249 (97%)
CE 16×16	5,809	3,618 (62%)	2,191 (38%)
WE 17×17	21,569	2,437 (11%)	19,132 (89%)
Total	39,270	10,439 (27%)	28,831 (73%)

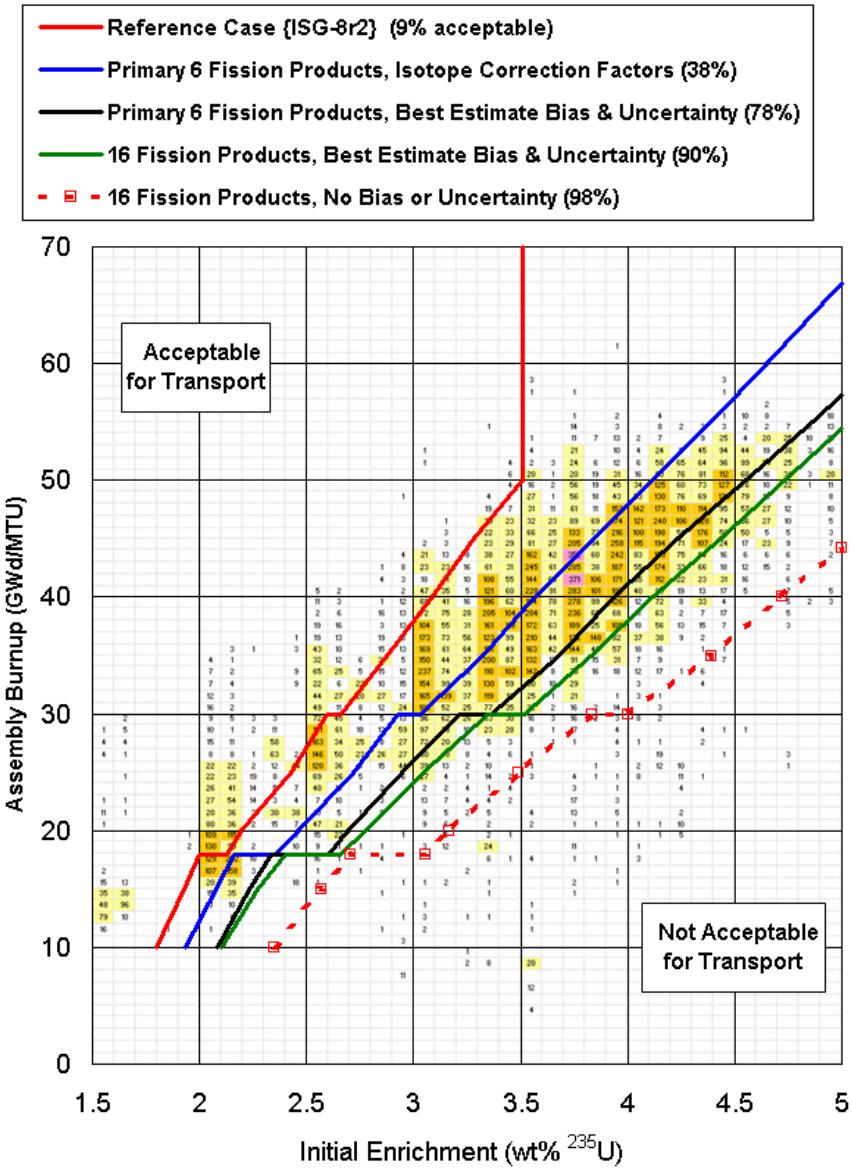

Figure 7. Comparison of calculational assumptions for WE 17 × 17 fuel assemblies. Percentages of inventory acceptable for the GBC-32 cask are shown in parentheses. "Primary 6 Fission Products" are ^{103}Ru, ^{133}Cs, ^{149}Sm, ^{143}Nd, ^{151}Sm, and ^{155}Gd . "16 Fission Products" are ^{95}Mo, ^{99}Tc, ^{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, and ^{155}Gd. *Source*: Ref. 28.

II.J Isotopic Validation

Uncertainties in the predicted isotopic concentrations in SNF represent one of the largest sources of overall uncertainty in criticality calculations that use burnup credit. As shown in Ref. 26, the methods used to propagate the uncertainties in the calculated nuclide concentrations to the uncertainty in the predicted k_{eff} of the system can have a significant effect on the uncertainty in the safety margin in criticality calculations and ultimately affect the potential capacity of spent fuel transport and storage casks employing burnup credit. Therefore, efforts were initiated to investigate, compare, and document [29] approaches for considering the effects of nuclide uncertainties in burnup-credit analyses. The potential

benefits of the various strategies were assessed based on the GBC-32 cask design. The subcritical margin estimated using best-estimate methods was compared with the margin estimated using conventional bounding methods [i.e., isotopic correction factors (ICFs)] of uncertainty propagation. To quantify the comparison, each of the strategies for estimating uncertainty was performed using a common database of spent fuel isotopic assay measurements (i.e., destructive measurements of isotopic compositions) for PWR fuels and predicted nuclide concentrations obtained using the SCALE code system. The experimental database applied in the study was expanded, from that used in previous studies, to include 56 spent fuel assay samples that included the important burnup-credit actinides and some limited fission product measurements.

The study demonstrated that the bounding method (ICFs), while easy to implement and clearly easy to defend as conservative, results in limiting and unrealistically large margins to account for nuclide variability. For actinide-only burnup credit calculations, the margins for nuclide uncertainty predicted using best-estimate methods were about one half the limiting margins predicted using the bounding method. When fission products were included in the analysis, the benefits of the best-estimate methods were even larger. The method of propagating the effects of nuclide uncertainties in the bounding method overestimates the impact of nuclide uncertainties on the predicted k_{eff} and precludes a realistic (statistically based) evaluation of the real subcritical margin. In contrast, the different best-estimate strategies for combining and propagating nuclide uncertainties were demonstrated to enable more accurate estimate of the effects of nuclide uncertainty by realistically representing the effects of variability in the nuclide concentrations. The effects of nuclide uncertainty were addressed separately from the bias, which is a nonrandom systematic error. However, both components must ultimately be considered in determining an appropriately conservative margin of subcriticality in a criticality calculation.

Efforts following the work of Ref. 29 have primarily focused on the acquisition, evaluation, and analysis of additional destructive radiochemical assay measurement data, with emphasis on recently available high enrichment and high burnup data that include fission product measurements [30, 31, 32].

II.K Commercial Reactor Critical Configurations for Criticality Validation

To address questions that arose related to the use of commercial reactor critical (CRC) state points for criticality validation, a study [33, 34] was performed to assess the neutronic similarities that may exist between a generic cask containing typical SNF assemblies and CRC state points. Forty CRC state points from five PWRs were selected for the study, and the type of CRC state points that may be applicable for validation of burnup credit criticality safety calculations for spent fuel transport, storage, and disposal systems were identified. The study employed the cross-section sensitivity and uncertainty analysis methods [35] and tools in the SCALE code system as a means to investigate system similarity on an integral and nuclide-reaction-specific level. The results indicated that, except for the fresh fuel core configuration, all analyzed CRC state points are either highly similar, similar, or marginally similar to the GBC-32 cask containing SNF assemblies with burnups ranging from 10 to 60 GWd/MTU. Based on the integral system parameter, c_k , [35] approximately 30 of the 40 CRC state points were identified as applicable to validation of burnup credit in the GBC-32 cask containing typical spent fuel assemblies with burnups ranging from 10 to 60 GWd/MTU. The state points providing the highest similarity ($c_k > 0.95$) were attained at or near the end of a reactor cycle. The c_k values were dominated by neutron reactions with major actinides and hydrogen, as the sensitivities of these reactions are much higher than those of the minor actinides and fission products. On a nuclide-reaction-specific level, the CRC state points were found to provide significant similarity for most of the actinides and fission products relevant to burnup credit. A comparison of energy-dependent sensitivity profiles showed a slight shift of the CRC k_{eff} sensitivity profiles toward higher energies in the thermal region as compared to the k_{eff} sensitivity profile of the GBC-32 cask. Parameters representing coverage of the application by the CRCs on an

energy-dependent, nuclide-reaction-specific level (i.e., effectiveness of the CRCs for validating the cross sections as used in the application) were also examined. Based on the CRCs with $c_k > 0.8$ and an assumed relative standard deviation for uncovered covariance data of 25%, the relative standard deviation of k_{eff} due to uncovered sensitivity data was found to vary from 0.79% to 0.95% for cask burnups ranging from 10 to 60 GWd/MTU. As expected, this uncertainty in k_{eff} was largely dominated by noncoverage of sensitivities from major actinides and hydrogen. The contributions from fission products and minor actinides were very small and comparable to statistical uncertainties in k_{eff} results. These results (again, assuming a 25% uncertainty for uncovered covariance data) indicate that there could be approximately 1% uncertainty in the calculated application k_{eff} due to incomplete neutronic testing (validation) of the software by the CRCs. However, this conclusion also assumes that all other uncertainties in the complex CRC configurations (e.g., fuel isotopic compositions, physical characteristics of reactor core components, and reactor operating history information) are well known. An evaluation and a quantification of the uncertainties in the CRC configurations were recommended prior to the use of CRCs for code validation (i.e., quantifying code bias and bias uncertainty).

II.L Fuel Assembly Misloading

The ISG8R2 recommends a burnup measurement for each assembly to confirm the reactor record and compliance with the assembly burnup value used for loading acceptance. This recommendation is intended to prevent unauthorized loading (misloading) of assemblies due to inaccuracies in reactor burnup records and/or improper assembly identification, thereby ensuring that the appropriate subcritical margin is maintained. To understand the significance, and corresponding diligence with which such misloadings should be prevented, it is necessary to understand the consequences of potential assembly misloading on the system k_{eff} value and to evaluate the associated increases in k_{eff} against inherent margins (e.g., an administrative margin), where present. To support this understanding, a study [36] was performed to determine the changes in k_{eff} that can result from a wide variety of postulated fuel misloading events in a representative high-capacity transport cask (i.e., the GBC-32 cask). The purpose of the study was to provide quantitative information for the impact of such events on the subcritical margin and aid in assessing the appropriate role for burnup measurements in ensuring safety. The report did not address the likelihood of occurrence for any of the misload configurations considered. The discussion that follows summarizes the analysis and findings in Ref. 36.

The study considered two representative burnup points on cask loading curves developed with and without the principal fission products present and included an investigation of (1) the effect of misloading fuel that had burnup values that were 90, 80, 50, 25, 10, and 0% of the minimum value required by the loading curve (i.e., underburned); (2) the effect of misloading fresh fuel with enrichment values of 2, 3, 4, and 5 wt % ^{235}U; and (3) for each scenario, the effect of misloading multiple (1, 2, 3, and 4) assemblies. For the representative, qualified burnup-enrichment combinations, with and without fission products included, misloading two assemblies that were underburned by 75% resulted in an increase in k_{eff} of 0.025−0.045, while misloading four assemblies that were underburned by 50% also resulted in an increase in k_{eff} of 0.025−0.045. For the cask and conditions considered, a reduction in burnup of 20% in all assemblies was shown to result in an increase in k_{eff} of less than 0.035. Misloading a single fresh assembly with 3, 4, or 5 wt % ^{235}U enrichment resulted in an increase in k_{eff} of ~0.02, 0.04, or 0.06, respectively. Throughout the analysis, the misloaded assemblies were placed in the most reactive (central) positions within the cask. Therefore, the impact of misloading assemblies into noncentral positions (e.g., nearer to the radial cask periphery) is bounded by the cases considered.

In summary, the consequences to k_{eff} of loading assemblies that have slightly reduced burnup (e.g., 5% due to uncertainties in the burnup verification process), as compared with the required burnup, were fairly small (≤1%). On the other hand, loading one or more highly enriched (i.e., >4 wt %) fresh fuel assemblies has a significant consequence on criticality safety. These findings suggest that while it may not be

necessary to precisely verify the burnup value, it is necessary to ensure that fresh or very low burnup (nearly fresh) fuel assemblies are not misloaded into a cask.

II.M Criticality Validation

One of the most significant challenges associated with expanded implementation of burnup credit is the validation of criticality calculations used in the safety evaluation—in particular, the availability and use of applicable critical experiment data, particularly for fission products. Applicants and regulatory reviewers have been constrained by both a scarcity of data and a lack of clear technical bases (e.g., criteria) for demonstrating applicability of the data. NRC staff have noted that the rationale for restricting ISG8R2 to actinide-only is based largely on the lack of clear, definitive experiments that can be used to estimate the bias and uncertainty for computational analyses associated with using burnup credit. In fact, at the time ISG8 was originally issued, mixed-oxide (MOX) critical experiments with the same relative composition of uranium and plutonium nuclides found in typical SNF were not available. However, the fact that the neutron-absorbing properties of fission products are known to reduce the k_{eff} value beyond the actinide-only assumption formed the basis for the position that the actinide-only assumption was conservative [7].

To address the issue of criticality validation, efforts were directed to obtain, and make available to industry, a well-qualified experimental database that could ensure reliable and accurate estimation of any bias and uncertainty resulting from the codes and data used to predict the k_{eff}. Rather than an a priori decision on suitability of candidate experiments, ORNL sought to obtain and assess critical experiment data from the following sources:

1. critical experiments within the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (IHECSBE);
2. proprietary critical experiment data;
3. commercial reactor criticals (CRCs), i.e., critical state points from operating reactors; and
4. proposed new critical experiments.

However, these experiments do not contain the same relative compositions, or even all of the nuclides that are present in spent fuel, or have other aspects that impair their use in validation. Thus, there was a need to limit the nuclides to those present in the critical experiments (typically plutonium and uranium isotopes only) and/or demonstrate their applicability to a spent fuel inventory.

As part of this effort, ORNL negotiated to gain access to a series of proprietary critical experiments, referred to as the Haut Taux de Combustion (HTC) experiments, performed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) at their Valduc critical experiment facility. These experiments were of interest because the plutonium-to-uranium ratio and the isotopic compositions of both the uranium and plutonium used in the simulated fuel rods were designed to be similar to those of a typical PWR fuel assembly with an initial enrichment of 4.5 wt % ^{235}U and burnup of 37,500 MWd/MTU. The fuel material also includes ^{241}Am, which is present due to the decay of ^{241}Pu. The HTC experiments include configurations designed to simulate fuel handling activities, pool storage, and transport in casks constructed of thick lead or steel. A report [37] was prepared that discusses the evaluation of the four HTC experimental data reports, modeling of the experiments, sensitivity and uncertainty analysis, and upper subcritical limit (USL) calculation. The report also presents some conclusions and recommendations concerning use of the HTC experiment data for burnup credit applications. The similarity of the HTC experiments with PWR SNF was quantified using sensitivity/uncertainty analysis, confirming that the HTC experiments are significantly more applicable to the validation of burnup credit calculations than other available MOX experiments [37, 38]. It was noted that the HTC experiments were designed and executed with a high level of rigor, resulting in experimental uncertainties that are lower

than many of the earlier MOX experiments. The HTC data reports, together with information provided in the report, provide sufficient data to allow either detailed or simplified computational models to be developed. The HTC experiments substantially strengthen the technical basis for validation by adding to the previously small number of applicable experiments against which to compare burnup credit applications. Existing MOX experiments found applicable were also identified in the report [37].

Efforts following the work of Ref. 37 have primarily focused on the acquisition, evaluation, and analysis of additional critical experiment data and methods development related to the use of critical experiment data, with emphasis on data for fission products with high reactivity worth [39]. A principal challenge for crediting fission products in a burnup credit safety evaluation is the limited availability of relevant fission product critical experiments for bias and bias uncertainty determination. A recent paper [39] provides an evaluation of the available critical experiments that include fission products, along with bounding, burnup-dependent estimates of fission product biases generated by combining energy-dependent sensitivity data for a typical burnup credit application with the nuclear data uncertainty information distributed with SCALE 6. The paper describes a method for determining separate bias and bias uncertainty values for individual fission products and provides illustrative results. Additionally, a bias calculation method based on data adjustment techniques and reactivity sensitivity coefficients calculated with the SCALE sensitivity/uncertainty tools and some typical results are provided. Using the methods described in this Ref. 39, the bias for a representative high-capacity spent fuel cask (GBC-32) associated with the 16 most important stable or near-stable fission products is predicted to be no greater than 2% of the total worth of the 16 FPs, or <1.3% in k_{eff}.

II.N Burnup Measurements

As mentioned in Section II.L, ISG8R2 recommends an out-of-core burnup measurement to confirm the reactor record and compliance with the assembly burnup value used for cask loading acceptance. The recommendation is intended to prevent unauthorized loading (i.e., misloading) of assemblies due to inaccuracies in reactor burnup records and/or improper assembly identification, thereby ensuring that the appropriate subcritical margin is maintained. In response to industry feedback related to this recommendation, NRC initiated an effort to reevaluate this recommendation and evaluate potential alternatives to confirmatory burnup measurements [40]. In support of this effort, a report [41] was prepared to review and summarize information and issues relevant to preshipment burnup measurements when using burnup credit in PWR SNF transport and storage casks. In particular, the report reviewed the role of burnup measurements in the regulatory guidance for demonstrating compliance with burnup loading criteria, burnup measurement capabilities and experience, generation and accuracy of utility burnup records, fuel movement and misloading experience, and the consequences of misloading assemblies in casks designed for burnup credit [41].

III. RECENT ACTIVITIES AND FUTURE DIRECTIONS

Recent and current efforts are directed toward developing the technical basis and information for revising ISG8R2 to allow credit for fission products. As mentioned, these efforts are primarily focused on the acquisition, evaluation, and analysis of additional critical experiment and radiochemical assay data and on methods development related to the use of these data to support credit for fission products. The goal is to develop and establish a technically sound validation approach (both depletion and criticality) for SNF criticality safety evaluations based on best-available data and methods and to apply the approach for representative spent fuel storage and transport configurations/conditions to demonstrate its usage and applicability, as well as to provide reference bias results.

For isotopic validation, the planned approach is to determine burnup-dependent reactivity bias and bias uncertainty due to bias and bias uncertainty in isotopic predictions via comparisons of isotopic

composition predictions (calculated) and measured isotopic compositions from destructive radiochemical assay, utilizing as much assay data as is available, and a best-estimate Monte Carlo–based method [29]. For criticality validation, the planned approach is to utilize available critical experiment data, e.g., the HTC data, for validation of principal actinides; utilize as much available fission product critical experiment data as is possible; utilize calculated sensitivities and nuclear data uncertainties to predict individual biases for all relevant fission products as a function of burnup [39]; and verify predictions of biases based on sensitivities and nuclear data uncertainties with calculated biases based on the limited available fission product critical experiment data. These activities are ongoing and will be the subject of future papers.

IV. ACKNOWLEDGMENTS

The research reviewed in this paper was performed under contract with the U.S. Nuclear Regulatory Commission. The authors gratefully acknowledge W. J. Marshall, G. Radulescu, and J. M. Scaglione for their review and suggestions on this paper. Oak Ridge National Laboratory is managed by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy.

V. REFERENCES

1. *RW-859 Nuclear Fuel Data*, Energy Information Administration, Washington, D.C. (Oct. 2004).
2. "Spent Fuel Project Office Interim Staff Guidance – 8, Rev. 1—Limited Burnup Credit," U.S. Nuclear Regulatory Commission (July 30, 1999).
3. D. E. Carlson, C. J. Withee and C. V. Parks, "Spent Fuel Burnup Credit in Casks: An NRC Perspective," *Proc. U.S. Nuclear Regulatory Commission's Twenty-Seventh Water Reactor Safety Information Meeting*, October 25–27, 1999, Bethesda, Maryland, NUREG/CP-0169, 419–436 (2000).
4. *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel—Final Report*, NUREG-1617, U.S. Nuclear Regulatory Commission (March 2000).
5. *Standard Review Plan for Spent Fuel Dry Storage Facilities—Final Report*, NUREG-1567, U.S. Nuclear Regulatory Commission (March 2000).
6. "Spent Fuel Project Office Interim Staff Guidance – 8, Rev. 2—Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," U.S. Nuclear Regulatory Commission (September 27, 2002).
7. C. V. Parks, C. J. Withee, and J. C. Wagner, "U.S. Regulatory Recommendations for Actinide-Only Burnup Credit in Transport and Storage Casks," *Proc. IAEA Technical Meeting on Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition*, August 29–September 2, 2005, London, U.K., IAEA-TECDOC-1547, ISBN 92-0-103307-9, Date of Issue: June 21, 2007.
8. C. V. Parks, M. D. DeHart, and J. C. Wagner, *Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel,* NUREG/CR-6665 (ORNL/TM-1999/303), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, February 2000.
9. C. V. Parks, I. C. Gauld, J. C. Wagner, B. L. Broadhead, M. D. DeHart, and D. D. Ebert, "Research Supporting Implementation of Burnup Credit in the Criticality Safety Assessment of Transport and Storage Casks," in *U.S. Nuclear Regulatory Commission Proc. Twenty-Eighth Water Reactor Safety Information Meeting,* Bethesda, MD, October 23–25, 2000.
10. G. H. Bidinger et al., *Burnup Credit PIRT Report,* NUREG/CR-6764 (BNL-NUREG-52654), prepared for the U.S. Nuclear Regulatory Commission by Brookhaven National Laboratory, Upton, NY, May 2002.
11. J. C. Wagner and C. V. Parks, *Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage,* NUREG/CR-6683 (ORNL/TM-2000/230), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, September 2000.

12. United States Nuclear Regulatory Commission, Title 10 Code of Federal Regulations (CFR), Part 50, Section 68, Criticality accident requirements.
13. J. C. Wagner and C. V. Parks, "Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage," *Nucl. Technol*. **136(1)**, 130–140, October 2001.
14. J. C. Wagner, *Computational Benchmark for the Estimation of the Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit,* NUREG/CR-6747 (ORNL/TM-2000/306), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, August 2001.
15. I. C. Gauld and S. M. Bowman, *STARBUCS: A Prototypic SCALE Control Module for Automated Criticality Safety Analyses Using Burnup Credit*, NUREG/CR-6748 (ORNL/TM-2001/33), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, October 2001.
16. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, ORNL/TM-2005/39, Version 6, Vols. I–III, Oak Ridge National Laboratory, Oak Ridge, TN, January 2009. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-750.
17. G. Radulescu and I. C. Gauld, "Enhancements to the Burnup Credit Criticality Safety Analysis Sequence in SCALE," *Proc. 2009 Nuclear Criticality Safety Division Topical Meeting on Realism, Robustness and the Nuclear Renaissance*, Richland, WA, September 13-17, 2009.
18. J. C. Wagner and C. V. Parks, *Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit,* NUREG/CR-6761 (ORNL/TM-2000/373), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, March 2002.
19. C. E. Sanders and J. C. Wagner*, Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit,* NUREG/CR-6760 (ORNL/TM-2000/321), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, March 2002.
20. J. C. Wagner and C. E. Sanders, "Investigation of the Effect of Fixed Absorbers on the Reactivity of PWR Spent Nuclear Fuel for Burnup Credit," *Nucl. Technol*. **139(2)**, 91–126, August 2002.
21. C. E. Sanders and J. C. Wagner, *Parametric Study of the Effect of Control Rods for PWR Burnup Credit*, NUREG/CR-6759 (ORNL/TM-2001/69), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, February 2002.
22. J. C. Wagner and C. V. Parks, *Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses*, NUREG/CR-6781 (ORNL/TM-2001/272), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, January 2003.
23. J. C. Wagner and M. D. DeHart, *Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations*, ORNL/TM-1999/246, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, Oak Ridge, TN, March 2000.
24. R. J. Cacciapouti, 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.
25. 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), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, March 2003.
26. J. C. Wagner and C. E. Sanders, *Assessment of Reactivity Margins and Loading Curves for PWR Burnup Credit Cask Designs*, NUREG/CR-6800 (ORNL/TM-2002/6), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn., March 2003.
27. J. C. Wagner, "Evaluation of Burnup Credit for Accommodating PWR Spent Nuclear Fuel in High-Capacity Cask Designs," in *Proc. of the 7th International Conference on Nuclear Criticality Safety (ICNC2003)*, pp. 684–689, Tokai-mura, Japan, October 20–24, 2003.

28. J. C. Wagner and D. E. Mueller, "Updated Evaluation of Burnup Credit for Accommodating PWR Spent Nuclear Fuel to High-Capacity Cask Designs," presented at the 2005 NCSD Topical Meeting, Knoxville, TN, September 19–22, 2005.
29. I. C. Gauld, *Strategies for Application of Isotopic Uncertainties in Burnup Credit*, NUREG/CR-6811 (ORNL/TM-2001/257), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, June 2003.
30. G. Ilas, I. C. Gauld, F. C. Difilippo, and M. B. Emmett, *Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors*, NUREG/CR-6968 (ORNL/TM-2008/071), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, February 2010.
31. G. Ilas, I. C. Gauld, and B. D. Murphy, *Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—ARIANE and REBUS Programs UO_2 Fuel*), NUREG/CR-6969 (ORNL/TM-2008/072), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, February 2010.
32. I. C. Gauld, G. Radulescu, and G. Ilas, "SCALE Validation Experience Using an Expanded Isotopic Assay Database for Spent Nuclear Fuel," *Proc. IAEA/CSN International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing, and Disposition*, Cordoba, Spain, October 27–30, 2009.
33. G. Radulescu, D. E. Mueller, and J. C. Wagner, *Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit*, NUREG/CR-6951 (ORNL/TM-2006/87), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, January 2008.
34. G. Radulescu, D. E. Mueller, and J. C. Wagner, "Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit," *Nucl. Technol.* **167(2)**, 268–287, August 2009.
35. B. L. Broadhead, B. T. Rearden, C. M. Hopper, J. J. Wagschal, and C. V. Parks, "Sensitivity- and Uncertainty-Based Criticality Safety Validation Techniques," *Nucl. Sci. Eng.* **146**, 340 (2004).
36. J. C. Wagner, *Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*, NUREG/CR-6955 (ORNL/TM-2004/52), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, January 2008.
37. D. E. Mueller, K. R. Elam, and P. B. Fox, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*, NUREG/CR-6979 (ORNL/TM-2007/083), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, September 2008.
38. D. E. Mueller and J. C. Wagner, "Application of Sensitivity/Uncertainty Methods to Burnup Credit Criticality Validation," presented at the IAEA Technical Meeting on Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition, London, U.K., August 29–September 2, 2005.
39. D. E. Mueller, B. T. Rearden and D. A. Reed, "Evaluation of Fission Product Critical Experiments and Associated Biases For Burnup Credit Validation, *Proc. IAEA/CSN International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing, and Disposition,* Cordoba, Spain, October 27–30, 2009.
40. Andrew Barto and Natreon Jordan, "Regulatory Perspective on Confirmatory Burnup Measurements for Burnup Credit in Spent Nuclear Fuel Transportation Packages," *Proc. IAEA/CSN International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing, and Disposition*, Cordoba, Spain, October 27–30, 2009.
41. B. B. Bevard, J. C. Wagner, C. V. Parks, and M. Aissa, *Review of Information for Spent Nuclear Fuel Burnup Confirmation*, NUREG/CR-6998, prepared for the Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, TN, December 2009.