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Phenomena and Parameters Important to Burnup Credit

C. V. Parks, M. D. DeHart, and J. C. Wagner

Nuclear Engineering Applications Section
Oak Ridge National Laboratory
P. O. Box 2008
Oak Ridge, TN 37831-6370
865 (574-5280)
parkscv@ornll.gov

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PHENOMENA AND PARAMETERS IMPORTANT TO BURNUP CREDIT

C. V. PARKS, M. D. DEHART, J. C. WAGNER
Oak Ridge National Laboratory (ORNL),
Oak Ridge, Tennessee
USA

Abstract

Since the mid-1980s, a significant number of studies have been directed at understanding the phenomena and parameters important to implementation of burnup credit in out-of-reactor applications involving pressurized-water-reactor (PWR) spent fuel. The efforts directed at burnup credit involving boiling-water-reactor (BWR) spent fuel have been more limited. This paper reviews the knowledge and experience gained from work performed in the United States and other countries in the study of burnup credit. Relevant physics and analysis phenomenon are identified, and an assessment of their importance to burnup credit implementation for transport and dry cask storage is given.

1. INTRODUCTION

In contrast to criticality safety analyses that employ the fresh-fuel assumption, credit for fuel burnup necessitates careful consideration of the fuel operating history, additional validation of calculational methods (due to prediction and use of nuclide compositions for spent fuel), consideration of new conditions or configurations for the licensing basis, and additional measures to ensure proper cask loading. For pressurized-water-reactor (PWR) fuel, each of these four areas have been studied in some detail over the last decade and considerable progress has been made in understanding the issues and developing the information needed for an effective safety evaluation that applies burnup credit. More recently, studies to expand the understanding needed to use burnup credit with spent nuclear fuel (SNF) from boiling-water reactors (BWRs) have been performed in the United States. The purpose of this paper is to identify the characteristic parameters and physics phenomena that are important to understanding burnup credit and review the current knowledge as gleaned from the studies performed in the United States, in other countries, and within international organizations. The following sections discuss the parameters and physics associated with the nuclides important to burnup credit, depletion and decay phenomena, and modeling of a SNF cask.

2. NUCLIDES IMPORTANT TO BURNUP CREDIT

Spent nuclear fuel contains hundreds of unique nuclides. The actual reactivity worth of the fuel is a function of the net neutron production and absorption by all nuclides present. However, if criticality calculations are performed based on all fissile nuclides and a limited subset of absorbers, the calculated value of the effective neutron multiplication factor (k_{eff}) is conservative (i.e., k_{eff} is overestimated). To date, the approach proposed in the United States for burnup credit in storage and transport casks has involved the qualification of calculated isotopic predictions via validation against destructive assay measurements from SNF samples. Thus, utilization of nuclides in the safety analysis process has been limited based on the availability of measured assay data and chemical characteristics (e.g., volatility) that might cause the nuclide to escape the fuel matrix [1,2].

Several studies have been performed to identify the nuclides that have the most significant effect on the calculated value of k_{eff} as a function of burnup and cooling time [2,3]. Figures 1–3 provide the results of one study [3] which performed a relative ranking based on the fraction of total absorptions for each nuclide. The adequacy of this simple ranking approach has been confirmed with more rigorous approaches that obtained the actual change in k_{eff} for an infinite lattice of rods based on a change in each nuclide [2]. The relative worth of the nuclides will vary somewhat with fuel design, initial enrichment, and cooling time, but

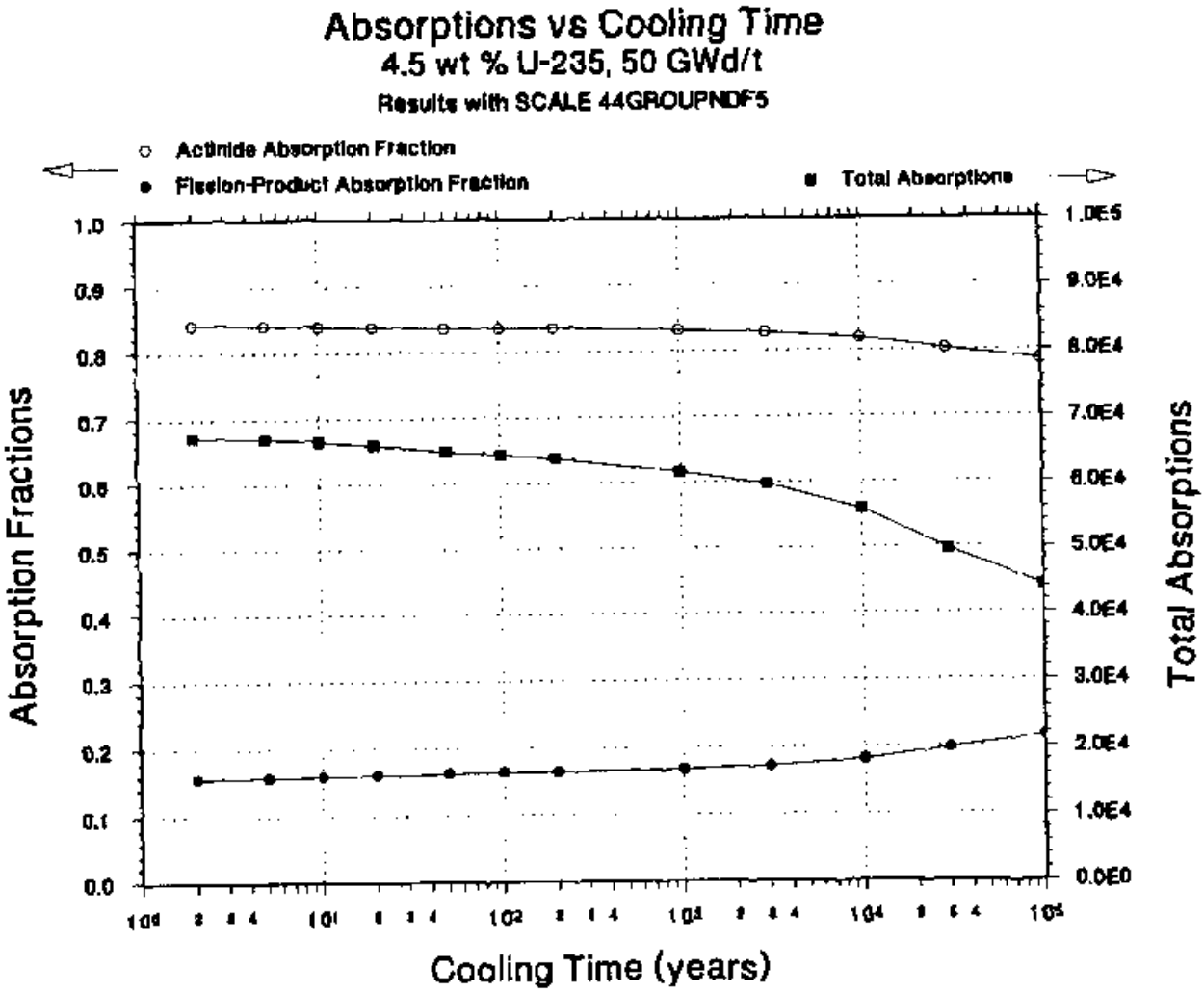

FIG. 1. Fraction of Neutron Absorptions versus Cooling Time for 4.5-wt %-Enriched PWR Fuel Burned to 50 GWd/t

the important nuclides remain the same. A recent study for BWR spent fuel also indicates the ranking of important nuclides changes only slightly in going from PWR to BWR operating conditions [4], and that the important nuclides are the same.

Figures 1–3 indicate that the majority of neutron absorption is caused by only a few actinide isotopes and, individually, the fission products contribute much less to neutron absorption. Within the cooling time range of interest to transport and dry cask storage (approximately 2 to 100 years), Figures 2–3 indicate that the relative importance of only a few nuclides change significantly. The buildup of ^{155}Gd and ^{147}Sm from the decay of other essentially non-absorbing fission products and the decay of ^{241}Pu (14.4 y-half-life) to ^{241}Am contribute to the decrease in k_{eff} as cooling time increases. The effect of the decay of ^{151}Sm appears to be compensated by the commensurate buildup of ^{151}Eu. Based on these and other studies, the nuclides listed in Table I are considered to be the prime candidates for inclusion in burnup credit analyses related to storage and transport casks. Obviously, ^{151}Sm (90-y half-life) and ^{151}Eu are a pair, and ^{151}Eu only needs to be considered if the absorption credit for ^{151}Sm must be maintained. Note, ^{135}Cs is a relatively minor absorber that has a negligible effect on cask reactivity; however, it has been included in many previous studies because measured isotopic data currently exist.

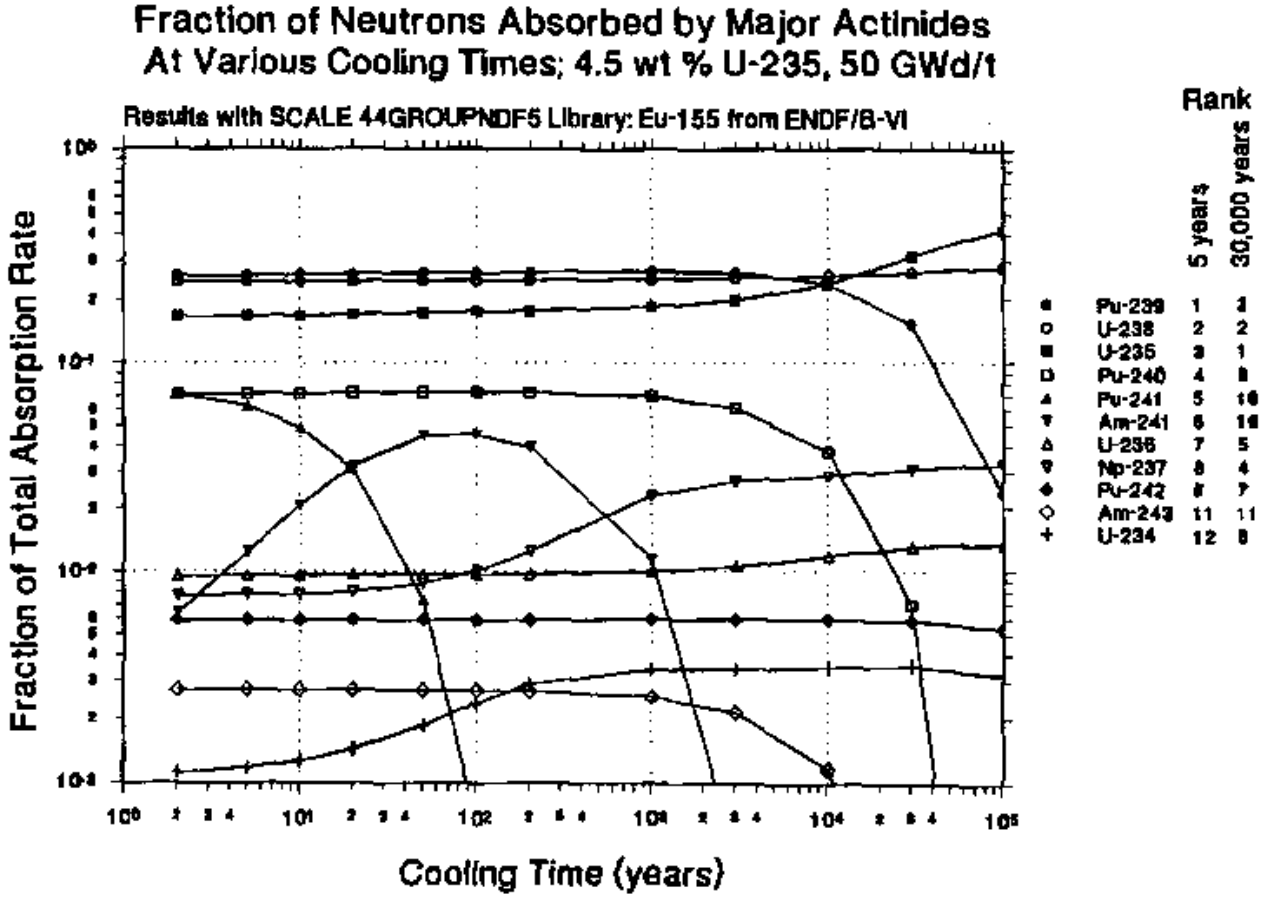

FIG. 2. Fraction of Neutron Absorbed by Major Actinides at Various Cooling Times for 4.5-wt %-Enriched PWR Fuel Burned to 50 GWd/t

As indicated earlier, validation of calculated isotopic predictions against experimental measurements is desirable for any nuclide upon which burnup credit criticality calculations are based. For BWR fuel, the number of nuclides for which there are measured data is significantly reduced and is limited primarily to the actinides of Table I [5]. For the most part, the fission product measurements available in the United States for PWR fuel is limited to 3–6 measurements, and prediction methods for these nuclides may not be considered to be fully validated [6]. This situation is a major reason that only partial or "actinide-only" burnup credit was considered in the U.S. Department of Energy (DOE) Topical Report on burnup credit [1] and the current U.S. regulatory guidance on burnup credit for transport and storage casks [7]. The fission product margin is still present, but since sufficient measured data for isotopic validation do not exist, credit for its negative worth has not been recommended for inclusion in safety analyses.

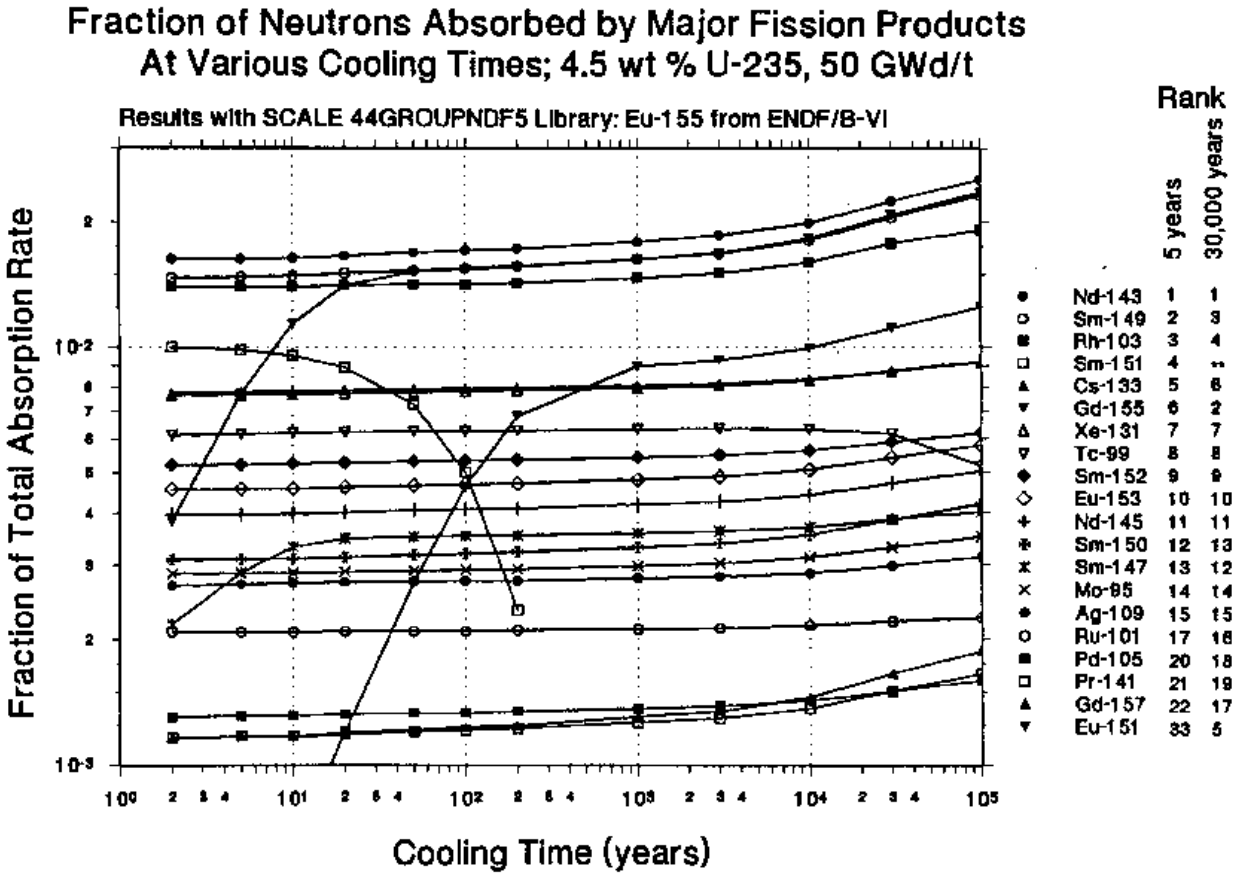

FIG. 3. Fraction of Neutrons Absorbed for Major Fission Products at Various Cooling Times for 4.5-wt %-Enriched PWR Fuel Burned to 50 GWd/t

TABLE I. PRIME CANDIDATE NUCLIDES FOR BURNUP CREDIT CRITICALITY CALCULATIONS

^{234}U	^{235}U	^{236}U	^{238}U	^{238}Pu	^{239}Pu
^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am*	^{237}Np
^{95}Mo*	^{99}Tc	^{101}Ru*	^{103}Rh*	^{109}Ag*	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{151}Eu*	^{152}Sm	^{153}Eu	^{155}Gd		

*Nuclides for which measured chemical assay data is not currently available in the United States (note a very limited amount of data is available for ^{243}Am) .

Table II shows the participant-averaged incremental worth of actinides and fission products as determined by the Working Group on Burnup Credit, an international group of experts organized by the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) [8]. This particular study, one of many performed by the Working Group, involved an infinite lattice of fuel pins with an initial enrichment of 3.6 wt % ^{235}U and nuclides nearly identical to those of Table I. The results of Table II indicate that, for these burnup values, the reactivity decrease is roughly 2/3 due to actinides, another 1/3 due to fission products. This finding is consistent with earlier work [9] for infinite lattices. However, it is important to remember that the competing effect of external absorbers in cask designs will change this ratio

for finite cask analysis resulting in the fission products with less relative worth. This reduced effect is seen in Figure 4, which is based on a generic rail cask design with 5-year cooled fuel. This figure shows the reactivity worth of the eleven actinides, with measured assay data as identified in Table I, in comparison to the additional worth that can be obtained from: fission products with measured assay data as identified in Table I, all the nuclides of Table I, and all nuclides (approximately 230) for which cross-section data are available in Version 5 of the U.S. Evaluated Nuclear Data Files (ENDF/B-V). The fission products provide approximately 1/4 of the total reactivity decrease for this particular cask design, somewhat lower than the 1/3 value seen for infinite lattices.

TABLE II. OECD PHASE IA ΔK VALUES (ACTINIDES ARE RELATIVE TO FRESH FUEL)

		30 GWd/t	40 GWd/t
1-year cooled	Actinides	0.1922	0.2492
	Fission products	0.1054	0.1248
	Total	0.2976	0.3740
5-year cooled	Actinides	0.2094	0.2721
	Fission products	0.1161	0.1417
	Total	0.3255	0.4138

3. PARAMETERS FOR DEPLETION ANALYSIS

It is anticipated that burnup credit will be applied for a wide variety of fuel types, each irradiated under a variety of reactor operating conditions (temperature, PWR boron concentration, BWR blade/fixed poison usage, etc.). If a cask design is intended to accept such a variety of fuel, assumptions that encompass the known variations must be employed in depletion calculations to ensure that the nuclide content of the fuel is conservatively represented. Several studies [2, 10 to 13] have been performed to assess the effect of depletion modeling assumptions on SNF nuclide predictions. In these parametric analyses, calculated nuclide concentrations were used to calculate k_{eff} for infinite SNF pin lattices and generic casks loaded with SNF. Trends in the neutron multiplication were then examined as a function of each parameter to determine the conservative direction (e.g., high temperature vs. low temperature) for that parameter, and the magnitude of the effect over a realistic operating range.

For each parameter studied in Refs. [2, 10 to 13], the sensitivities of the neutron multiplication to changes in the parameter increases with increasing burnup. Furthermore, with the exception of specific power/operating history effects, all of the trends discussed below are related to spectral hardening. Spectral hardening results in an increased production rate of plutonium from increased epithermal neutron capture in ^{238}U. Enhanced plutonium production and the concurrent diminished fission of ^{235}U due to increased plutonium fission have the effect of increasing the reactivity of the fuel at discharge and beyond. The exact mechanisms that result in spectral hardening for various operating conditions are discussed below. Unlike the other parameters, specific power and operating history effects are driven by the balance of the various equilibrium states of the nuclides present, as a function of power.

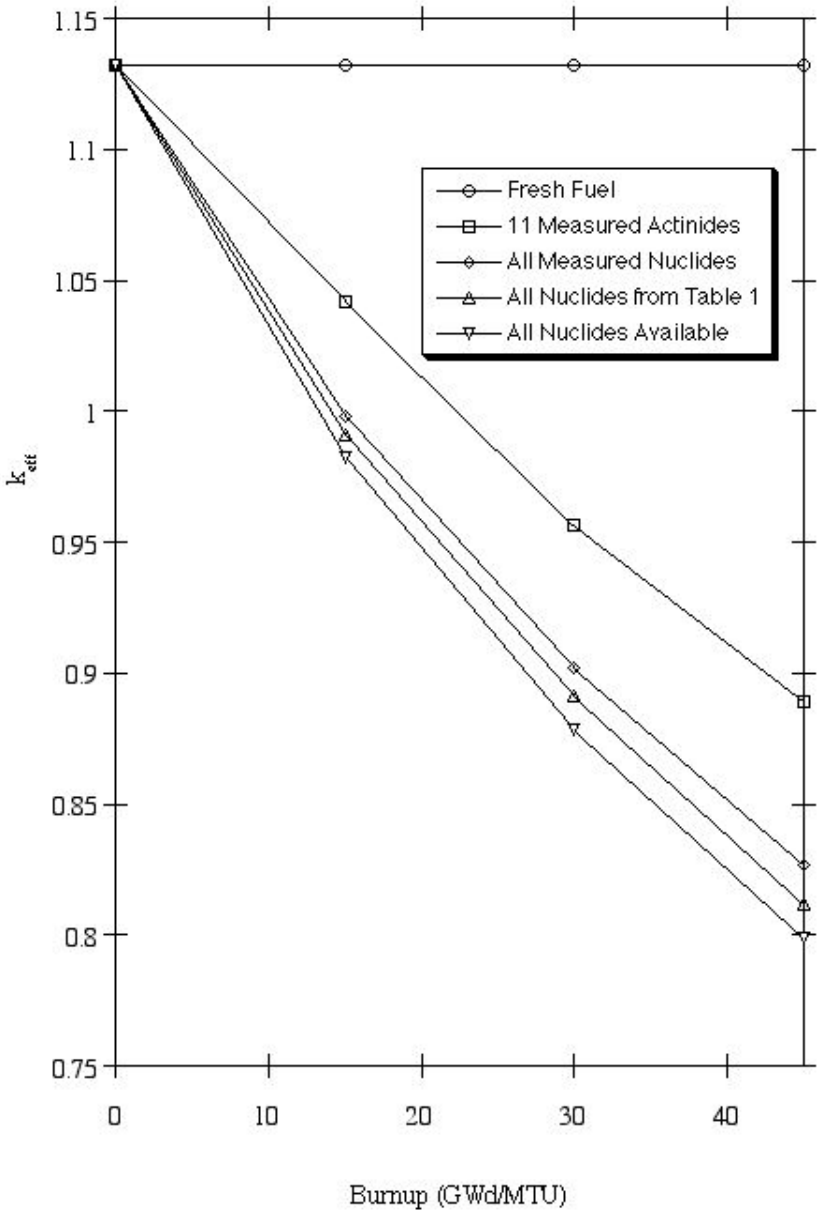

FIG. 4. Values of k_{eff} for a Generic Rail Cask as a Function of Burnup Using Different Sets of Isotopic Assumptions and 5-year Cooling Time

3.1. Fuel Temperature

Studies [2, 10 to 13] of both BWR and PWR depletion models show a clear trend for increased conservatism (i.e., increase in k_{eff} value) as the assumed fuel temperature during operation is increased. It is believed that at higher fuel temperatures, resonance absorption in ^{238}U is increased due to Doppler broadening, resulting in spectral hardening and increased plutonium production. The effect is burnup dependent, increasing linearly with increasing burnup. Thus conservative SNF nuclide inventories are predicted by assuming an upper estimate of fuel temperature during depletion calculations. The bounding case is for high-burnup fuel and Ref. [2] shows that the reactivity worth of temperature change is on the order of 5 pcm/K (pcm = percent mill = 10^{-5} $\Delta k/k$) for an infinite lattice of PWR fuel pins and 4 pcm/K for a generic cask [10]. Ref. [12] shows similar behavior for an infinite lattice of BWR fuel. Thus use of the maximum pellet-averaged temperature in the depletion analysis should be acceptable for PWR and BWR depletion analyses. A value of 1000 K would seem appropriately conservative to cover normal PWR reactor operations.

3.2. Moderator Temperature/Density

As with fuel temperatures, calculations performed with varying moderator temperatures show that nuclide compositions are most conservative with respect to neutron multiplication when calculated assuming an upper bound on moderator temperature (e.g., core outlet temperature) [2, 10 to 13]. Although the mechanisms are different, the net effect is the same. In a PWR, as the moderator temperature increases, the moderator density decreases. Decreased density results in reduced moderation, which results in spectral hardening. The response is close to linear, but has a slight exponential shape with increasing moderator density (due to the fact that water density is not linear with respect to temperature). The reactivity effect also increases with increasing burnup. For the bounding case of high-burnup fuel, Ref. [2] shows a reactivity worth of about 90 pcm/K for an infinite lattice of PWR fuel pins and Ref. [10] indicates 35 pcm/K in a cask environment. In general, however, the variation in temperature and corresponding density is relatively small in a PWR design. Thus, use of the maximum core outlet temperature (e.g., 600 K) is recommended.

Spectral hardening resulting from decreased moderator density is intentionally applied in the control of a BWR. However, the net effect is unchanged from the effect discussed for PWR designs. In BWR systems, moderator temperatures change very little axially once boiling begins in the flow channel. However, reactor operation allows significant variation in axial moderator density as the void fraction increases with increasing height. The void fraction can change significantly both axially and as a function of time. Hence, it is more instructive to study depletion effects as a function of moderator density rather than moderator temperature. Reference [12] demonstrates that for an infinite lattice of BWR assemblies, k_{inf} increases linearly with decreasing moderator density and the trend is more pronounced as the SNF burnup increases. The magnitude of the effect is on the order of 10^3 pcm/(g/cm^3) for high burnup fuel. Thus, the highest average void fraction (minimum average moderator density) would appear to be the simple bounding value to use for depletion analysis of BWR fuel. Since the reactivity of BWR fuel in a cask is driven by the fuel at the top of the assembly, it is anticipated that using the highest average void fraction should be a prudent, yet practical assumption for the safety analysis. However, additional work in this area may be warranted to substantiate the initial findings.

3.3 Soluble Boron

Soluble boron is used to control excess reactivity in PWRs. Soluble boron concentrations of 1000–1500 ppm boron are typical at beginning-of-cycle and decrease to 0–200 ppm at end-of-cycle. Depletion calculations may model the boron change in steps, or assume an average boron concentration for a full cycle. Studies have been performed to assess the effect of the soluble boron concentration used during depletion [2,10,13,14]. The results of these bounding high-burnup calculations show a clear linear increase in reactivity with increased boron concentration at a rate of approximately 3 pcm/ppm for an infinite lattice of pins and 3.5 pcm/ppm in a cask configuration. Again, although the mechanism is different from that which occurs in fuel and moderator temperature variations, the end result is the same. Spectral hardening results from the absorption of thermal neutrons in the moderator by the soluble poison. As with temperatures, the effect of higher boron concentrations is more significant with higher burnup values, since more conversion occurs over the fuel cycle. Use of an average cycle boron value of 750 ppm should be adequately bounding based on the studies performed; however, the establishment of a bounding value for the maximum average boron per cycle based on boron let-down curves would be informative.

3.4. Specific Power and Operating History

The effect of various operating histories (variations in specific power with time) on the reactivity of spent fuel has been studied for a limited set of hypothetical power histograms [2,10,12]. Rather than attempt to determine a real operating history that would bound all other operating histories, histograms were developed to represent the key aspects of operating histories (e.g., extended downtime early in life, extended downtime

late in cycle, high- power operation early in life, short intercycle downtimes, long intercycle downtimes, etc.). Results showed a wide variability in response due to the significantly different decay rates and equilibrium concentrations for the nuclides studied. In general, low-power operation near the end of cycle produces the highest reactivity due to decreased fission product inventory. However, the opposite is true when only actinides are considered for burnup credit — high-power operation is more conservative at end of life. Fission product worth is more sensitive to specific power than that of actinides; thus, when both are present, the net effect is driven by fission product behavior. Hence low-power operation toward end of life yields the most conservative estimate of reactivity. The net effect is rather small, up to 200 pcm for the operating histories considered. It appears that the optimum approach would be to assume a simple continuous-power operating history, and add in a margin to account for operating-history-induced effects.

The effect of specific power assumed during depletion calculations has also been studied independently of operating history for PWRs [2,10]. Although an operating history is simply a time-varying specific-power profile, it is important to understand the effect of the magnitude of specific power when held constant with time. Calculations with both actinide and fission product credit show a trend for conservative prediction of fuel reactivity worth when fuel is burned at lower specific power for a longer period of time for a given burnup. The magnitude of the conservatism increases with increasing burnup. However, the opposite is true for calculations in which only actinides are considered in criticality calculations. For actinide-only credit, higher specific powers result in the most conservative set of isotopics. The magnitude of the conservatism also increases with increasing burnup. The difference in behavior between actinides and fission products is due to the relatively short decay times of fission product precursors relative to actinides.

Recent work [12] has shown that for high-burnup fuel with fission products present, the behavior of the SNF neutron multiplication as a function of specific power departs from a linear response. For high-burnup fuel, the neutron multiplication initially increases with increasing specific power, before turning (e.g., in the range of 10–20 MW/t) and decreasing as specific power continues to increase. Thus, there is a specific power that maximizes the neutron multiplication for high-burnup fuel with actinides and fission products assumed. The phenomenon will require further study to understand and quantify.

3.5. Burnable Absorbers

Burnable absorbers may be classified into two distinct categories: (1) Burnable Poison Rods (BPRs) and (2) Integral Burnable Absorbers. BPRs are rods containing neutron absorbing material that are inserted into the guide tubes of a PWR assembly during normal operation and are commonly used for reactivity control and enhanced fuel utilization. Due to the depletion of the neutron absorbing material, BPRs are often (but not always) withdrawn after one-cycle residence in the core. In contrast to BPRs, integral burnable absorbers are burnable poisons that are a non-removable or integral part of the fuel assembly once it is manufactured. An example of an integral burnable absorber is the Westinghouse Integral Fuel Burnable Absorber (IFBA) rod, which has a coating of zirconium diboride (ZrB_2) on the fuel pellets.

The net effect of poison rods is the same as that of soluble boron, since the same mechanism applies: preferential removal of thermal neutrons. However, rod effects are more localized, resulting in localized spectral hardening, non-uniform burnup across the assembly at a given axial height, and atypical axial burnup profiles. Recently completed studies have demonstrated that assemblies exposed to BPRs will show an increased k_{eff} in the range of 0.5% to 3% Δk depending on the number and poison loading of BPRs present, the length of the exposure to BPRs, the initial fuel enrichment and the burnup of the assembly. Inclusion of the axial burnup distribution reduces the reactivity increase associated with the BPRs. This is due to the fact that the lower burnup regions near the ends, which control the reactivity of the fuel when the axial burnup distribution is included, have less burnup, and thus less than average burnup exposure to the BPRs.

Assuming maximum BPR exposure during depletion would be a simple, conservative approach to bound the reactivity effect of BPRs; where maximum BPR exposure may be defined as the maximum possible

number of BPRs with the most bounding BPR design (i.e., most bounding geometric design and maximum possible poison loading) for the entire depletion. Other less conservative approaches, incorporating risk-informed approaches regarding the percentage of assemblies exposed to mutiple cycles, will be explored in the future.

A study has recently been completed that investigated the impact of integral burnable absorbers on the k_{eff} values in cask environments. Depending on the design and loading of neutron poison, the presence of integral burnable absorbers can slightly lower or raise the k_{eff} values of SNF assemblies, in comparison to assemblies without the integral burnable absorber. Integral burnable absorber analyses for multiple designs have been studied, and the maximum increase in k_{eff} is less than that identified for assemblies depleted with BPRs present.

The impact of control rods used during power operations can also have the effect of increasing fissile plutonium production at the ends of the fuel. Parametric studies to understand the potential magnitude of these effects are planned for the near future.

3.6. Summary of Depletion Modeling Parameters

Table III summarizes the discussion in the preceding paragraphs, including specific power and operating history effects. No specific recommendations for bounding parameters are given. Although expected values are listed in the preceding subsections, these values should be confirmed or revised by a survey of operational data before firm recommendations are made. Simultaneous use of realistic bounding parameter values in a depletion model provides a simple, prudent approach to the modeling process since it is unlikely that any fuel would be depleted under all such conditions simultaneously. However, the use of bounding values and/or models may not be the most appropriate for a risk-informed approach to implementing burnup credit. Work to investigate more realistic approaches based on actual ranges of operating conditions and the statistical probability of "outlier" bounding conditions will be explored in the future. However, a key to the success of such approaches is development of a database that provides information on the range of actual operating conditions with sufficient data points that "typical" conditions can be established. A reference industry report establishing a defensible value for PWR and BWR operations would be beneficial to facilitate future safety analyses.

4. COOLING TIME

Fuel discharged from a reactor increases in reactivity for several days due to the decay of short-lived poisons. After this point, reactivity decreases continuously with time out to about 100 years, at which time it begins to increase. The reactivity continues to increase until a second peak at around 30,000 years, after which time it begins decreasing again [3]. The reactivity of the second peak is always less than that occurring at 5 years when actinide and fission products are used in the criticality analysis. This means that an assumed cooling time for 5 years is conservative for any cooling time beyond 5 years. The magnitude of the conservatism depends on the initial enrichment and burnup of the fuel [2,3].

The effect of cooling time on k_{eff} for an infinite PWR pin-cell lattice is shown in Fig. 5 for various burnup and initial enrichment values. Note that as burnup increases, the effect of cooling time is more pronounced due to the increased quantity of ^{241}Pu and fission products relative to the remaining inventory. Reference [3] provides a comparison of absorption fraction versus burnup and further illustrates this increase in the negative reactivity worth from ^{241}Pu decay and fission product absorption. Since the reactivity of low-burnup fuel at the ends of the SNF is rather insensitive to cooling time and the reactivity of higher burned fuel decreases significantly with cooling time, the relative reactivity worth of the ends will increase with cooling time.

TABLE III. SUMMARY OF INFORMATION ON DEPLETION MODELING PARAMETERS

Parameter	**Bounding condition**	**Estimated sensitivity**	**Recommended conservative value/model**
Fuel temperature	Highest temperature	4–5 pcm/K	Max. pellet-average temperature
Moderator temperature (PWR)	Highest temperature	35–90 pcm/K	Maximum core outlet temperature
Moderator density (BWR)	Lowest density	10^3 pcm/(g/cm^3)	Minimum channel outlet density
Soluble boron concentration	Highest concentration	3–3.5 pcm/ppm	Maximum cycle-averaged concentration
Operating history	High power late in life (actinide-only)	N/A	Assume simple operating history, with margin of 200 pcm or more
Specific power	High specific power (actinide-only)	N/A	High but credible specific power
Fixed/Integral burnable absorbers	Burnable absorbers present during depletion	0.5 - 3% Δk over full range	Maximum absorber loading used for full irradiation history.

5. AXIAL BURNUP PROFILES

5.1. Phenomena Associated with Axial Effects

The dynamics of reactor operation result in non-uniform axial-burnup profiles in fuel with any significant burnup. At beginning of life in a PWR, a near-cosine axial flux shape will begin depleting fuel near the axial center of a fuel assembly at a faster rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission product poisoning that occurs near the center. However, because of the relatively high leakage near the end of the fuel, burnup will drop off rapidly near the ends. Partial length absorbers or non-uniform axial enrichment loadings can further complicate the burnup profile. In a BWR, the same phenomena occur [12], but the burnup profile is also influenced by the significantly varying moderator density profile and by non-uniform axial loadings of burnable poison rods and uranium enrichment.

The most reactive region of spent fuel is toward the ends, where there is an optimum balance between increased reactivity due to lower burnup and increased leakage due to closer proximity to the fuel ends[2]. A fairly extensive review of axial burnup distribution issues that are important to burnup credit criticality safety analyses is presented in Ref. [15]. The fact that there is a difference between the k_{eff} value calculated assuming an axially varying burnup profile and that calculated assuming a uniform (flat) burnup profile (associated with the average assembly burnup value) has become known as the “end effect.”

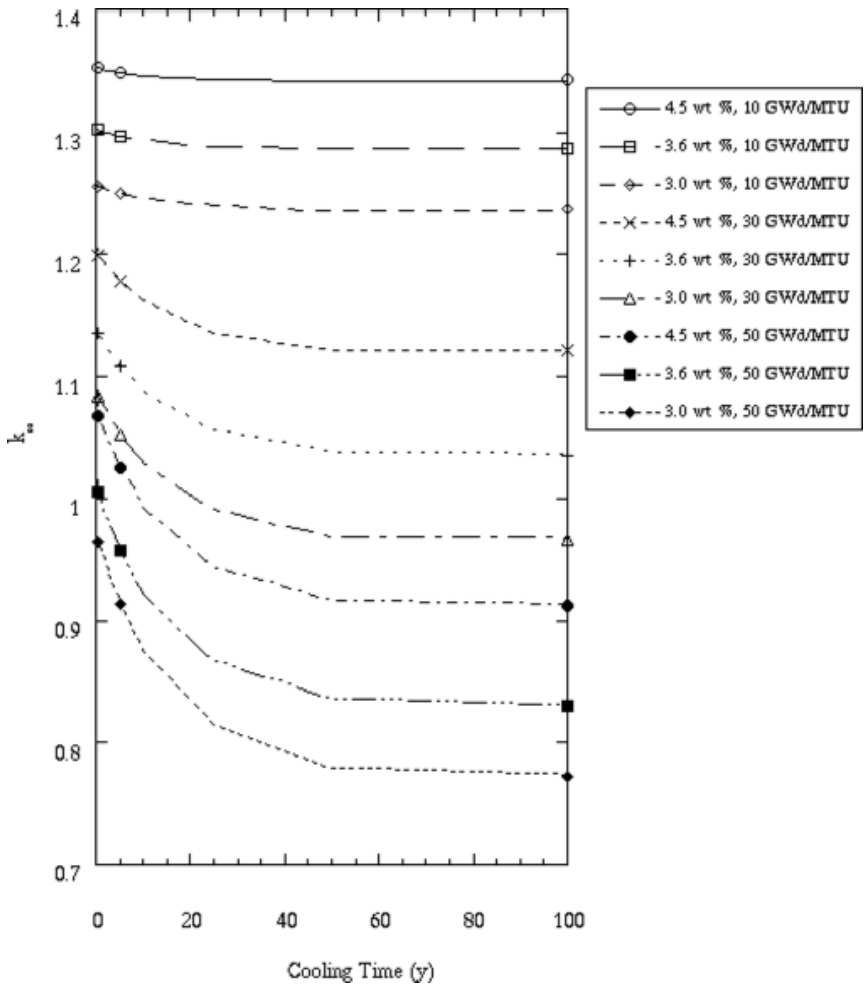

FIG. 5. Plot of k_{eff} versus Cooling Time for Various Enrichments and Burnup Values

Participants in the OECD/NEA Working Group on Burnup Credit performed criticality calculations for a 3-D infinite lattice of axially finite PWR pin cells [16]. The following items were noted in the results with respect to the end effect: (1) the end effect increases with increasing burnup and cooling time; (2) it is most pronounced when fission products are present; (3) the end effect is negative for low-burnup and short cooling times, but becomes positive and of greater magnitude at high-burnup and cooling time; (4) the cross-over from negative to positive occurs around 25 GWd/t when fission products are modeled, and near 30 GWd/t when fission products are not modeled; and (5) the crossover from negative to positive occurs at slightly higher burnup when fuel enrichment increases. In general, the same trends noted here for the infinite array model were also noted in the cask model analyzed by the participants [17].

In a BWR, the burnup profile is further complicated by several factors, including: (1) axially and time varying moderator density, (2) axially and radially varying fuel enrichments, (3) axially varying poison rod enrichments, and (4) partial control rod insertion. The reactivity of BWR fuel increases with burnup to a maximum or peak reactivity where the integral absorber (Gd) is nearly depleted. When considering the axial-burnup profile, it becomes apparent that local heights will not reach their peak reactivity simultaneously. Rather, the integral absorber will be depleted earlier near the axial center, and thus the reactivity will peak at the center while significant integral absorber is still present at the ends. Similar to PWR fuel, the axial burnup

distribution results in an increasing positive end effect with increasing burnup. However, early work [12,18] has shown the magnitude of the reactivity increase associated with the axial burnup distribution in BWRs may be larger than that which is typically observed for PWR fuel.

5.2. Profile Database

The true axial-burnup distribution is not known for the majority of spent fuel assemblies that will be loaded in a cask. In general, only the average burnup is known and documented in plant records associated with each SNF assembly. Thus to be conservative, one must identify and assume an axial-burnup profile that is realistic but is limiting in terms of the value of k_{eff} associated with the axially varying SNF nuclide compositions. To date, attempts to bound PWR profiles [2,10,11] in the United States have been based on a database of 3169 axial-burnup profiles for PWR assemblies [19]. The database of Ref. 19 consists of burnups calculated by utilities or vendors for a discrete number (18–24) of axial heights based on core-follow calculations and in-core measurement data. Although the profiles in the database are not measured directly, the use of the same analysis procedures for reactor core-following analyses inspires confidence that the profiles are representative of the actual fuel burnup.

If it is desirable to continue to base limiting axial profiles on profiles found to be limiting from a database, then it may be necessary to expand the existing database to include a broader variety of fuel designs, especially some of the more recent fuel designs. Furthermore, since control rods and partial-length absorbers can have a significant effect on axial profiles, a decision must be made whether to include or exclude such conditions in a database. Information on the use of control-rod insertion during normal reactor operations would be beneficial to better study and understand the potential impact on the axial profile and/or the SNF nuclide composition.

No attempt has been made to define a bounding profile for BWR fuel assemblies due to the lack of a database of burnup profiles. The fact that BWR fuel assemblies are manufactured with variable enrichments both radially and axially, are exposed to time-varying void distributions, contain fixed burnable poison rods, and are subject to partial control blade insertion during operation means that BWR profiles are likely to have more variation than that observed for PWR fuels. Thus a large database may be necessary to capture all of the important characteristics. Again, no such database exists for BWR profiles, and an industry activity to develop such a database would surely have value in implementation of burnup credit in cask storage and transport for BWR fuels.

5.3. Axial Modeling Approximations

In any spent fuel assembly, fuel burnup is a continuous function of axial location. However, in a numerical model, a depletion calculation must be performed for each finite burnup region in the model to estimate the contents of the spent fuel at that burnup state. Therefore, in practical application, spent fuel models must apply a set of discrete burnup intervals in which a constant burnup over each interval is assumed. As with any differencing approach, care must be taken to ensure that the spatial discretization is fine enough to capture physical phenomena. Sensitivity studies [2,10,12,15] have shown that a relatively coarse axial discretization, typically consisting of 7–11 axial regions, is sufficient to converge on the predicted eigenvalue for a spent fuel system. However, the axial discretization used in these studies and elsewhere [16,17] is non-uniform and tailored to the shape of the burnup profile. All known spent fuel profiles tend to be fairly uniform over most of the central region, but with significantly decreasing burnup near the axial ends of the active fuel. Thus, discrete models of burnup can use one to three burnup zones to represent the majority of the length of the fuel (central region), but more discrete zones are necessary to capture the more rapid change in burnup with position near the ends of the fuel. It would be valuable to safety analysts if there were criteria for determining the number and length of zones required in the model based on the axial profile being considered. An example of such criteria would be a zone for each 10% change in burnup. Such criteria need to be developed and tested.

As noted above, the spent fuel reactivity is a function of both the burnup distribution and axial leakage; thus the boundary conditions (i.e., assembly or cask conditions at the end of the fuel) may play a role in the strategy for determining appropriate axial modeling approximations. Calculations reported to date have been based on simple axial models with a fixed set of boundary conditions. Additional work may be needed to better evaluate potential limiting boundary conditions that should be used for normal and potential accident conditions.

6. HORIZONTAL BURNUP PROFILES

Radial variations in the neutron flux, which are primarily due to leakage at the core periphery, result in a non-uniform horizontal burnup distribution over the radial extent of the reactor core. As the reactor operates, the radial flux shape flattens due to fuel depletion and fission product poisoning near the core center. However, because of the high leakage near the core periphery, burnup drops off rapidly near the periphery. Ultimately, at the end of a cycle, the individual assemblies located near the center of the core will have a relatively uniform horizontal burnup distribution, while the assemblies near the core periphery may have a significant horizontal variation in burnup [20]. Fuel shuffling schemes designed to enhance fuel utilization typically relocate assemblies within the reactor core between cycle operations. These fuel management practices tend to effectively reduce the horizontal burnup gradient in normal discharged fuel. However, a periphery assembly discharged after a single irradiation cycle may exhibit a significant horizontal burnup gradient [20].

A database containing quadrant-wise horizontal burnup gradients for 808 PWR assemblies (Westinghouse 17×17 and Babcock and Wilcox 15×15) has been prepared [20], and the database has been examined for trends with the number of operating cycles, accumulated burnup, and initial enrichment. No trend with initial enrichment was observed. However, the horizontal gradient was shown to be inversely proportional to accumulated burnup and number of cycles, which are obviously closely related. In other words, the horizontal variation in burnup decreases with increasing burnup. Axial variation of the horizontal burnup distribution has not been addressed.

The horizontal variation in burnup is a criticality safety concern in the event that two or more assemblies are placed in a configuration such that their low-burnup regions are adjacent, thus resulting in an increase in reactivity [1]. This reactivity increase will be greatest in small cask designs — such as truck casks.

7. EPILOGUE

The basic information of this paper was derived from a report [21] prepared for the U.S. Nuclear Regulatory Commission (NRC) Office of Regulatory Research to help initiate a process called Phenomena Identification and Ranking Tables (PIRT), which has been used by the NRC to identify phenomena and prioritize their importance in helping to resolve a broad technical issue. This PIRT process involves the efforts of an international panel of experts. The final report from the PIRT process (due in 2001) will build on the foundation of Ref. [21] to provide an identification and ranking of phenomena and technology issues deemed important to effective burnup credit implementation and propose a table that prioritizes the areas where technical resolution is needed. The phenomena and technology issues, as well as the ranking table, will be updated by the NRC as additional input and feedback is obtained. Information on the PIRT process can be obtained at the following web site: http://www.nrc.gov/RES/pirt/BUC/index.html. Work continues at Oak Ridge National Laboratory to improve understanding and investigate analysis approaches that will facilitate safe implementation of burnup credit in transport and storage casks.

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REFERENCES

[1] Office of Civilian Radioactive Waste Management, *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages*, DOE/RW-0472, Rev. 2, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, September 1998.

[2] DEHART, M.D., *Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages*, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996.

[3] BROADHEAD, B.L., et al., *Investigation of Nuclide Importance to Functional Requirements Related to Transport and Long-Term Storage of LWR Spent Fuel*, ORNL/TM-12742, Lockheed Martin Energy Systems, Inc., Oak Ridge National Laboratory, June 1995.

[4] BROADHEAD, B.L., *K-infinite Trends with Burnup, Enrichment, and Cooling Time, for BWR Fuel Assemblies*, ORNL/M-6155, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, August 1998.

[5] HERMANN, O.W., DEHART, M.D., *Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel*, ORNL/TM-13315, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, September 1998.

[6] HERMANN, O.W., et al., *Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses*, ORNL/TM-12667, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, 1995.

[7] "Spent Fuel Project Office Interim Staff Guidance – 8, Rev. 1 – Limited Burnup Credit," USNRC, July 30, 1999. See also http://www.nrc.gov/OPA/reports/isg8r1.htm.

[8] TAKANO, M., *OECD/NEA Burnup Credit Criticality Benchmark —Result of Phase-1A*, JAERI-M 94-003 (NEA/NSC/DOC(93)22, Japan Atomic Energy Research Institute, 1994.

[9] PARKS, C.V., "Parametric Neutronic Analyses Related to Burnup Credit Cask Design," pp. 21–43 in *Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks*, Washington D.C., February 21–22, 1988, SAND89-0018, TTC-0884, UC-820, Sandia National Laboratory (October 1989).

[10] DEHART, M.D., *Parametric Analysis Of PWR Spent Fuel Depletion Parameters For Long-Term Disposal Criticality Safety*, ORNL/TM-1999/99, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, October 1999.

[11] KANG, C.H., AND LANCASTER, D.B., "Actinide-Only Burnup Credit for Pressurized Water Reactor Spent Nuclear Fuel – III: Bounding Treatment of Spatial Burnup Distributions," *Nucl. Technol*. **125**, 292 (1999).

[12] WAGNER, J.C., et al., *Investigation of Burnup Credit Modeling Issues Associated with BWR Fuel*, ORNL/TM-1999/193, UT-Battelle, LLC, Oak Ridge National Laboratory, October 2000.

[13] KANG, C.H., LANCASTER, D.B., *Depletion and Package Modeling Assumptions for Actinide-Only Burnup Credit*, DOE/RW-0495, U.S. Department of Energy (1997).

[14] THORNE, P.R., O'CONNER, G.J., BOWDE, R.L., "The Current Status of the Methods Adopted by BNFL in Claiming Credit for Burnup in Criticality Safety Assessments," *Sixth Int. Conf. on Nuclear Criticality Safety*, Versailles, France, September 20–24, 1999.

[15] WAGNER, J.C., DEHART, M.D., *Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations*, ORNL/TM-1999/246, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, March 2000.

[16] TAKANO, M., OKUNO, H., *OECD/NEA Burnup Credit Criticality Benchmark —Result of Phase IIA*, JAERI-Research-96-003 (NEA/NSC/DOC(61)01), Japan Atomic Energy Research Institute, 1996.

[17] NOURI, A., *OECD/NEA Burnup Credit Criticality Benchmark - Analysis of Phase II-B Results: Conceptual PWR Spent Fuel Transportation Cask*, IPS/98-05 (NEA/NSC/DOC(98)1), Instituted de Protection et de Surety Nuclear, May 1998.

[18] BROADHEAD, B.L., "Feasibility Assessment of Burnup Credit in the Criticality Analysis of Shipping Casks with Boiling Water Reactor Spent Fuel," *Nucl. Technol*. **110**, 1–21, April 1995.

[19] CACCIAPOUTI, R.J., VAN VOLKINBURG, S., *Axial Burnup Profile Database for Pressurized Water Reactors*, YAEC-1937, Yankee Atomic Electric Company, May 1997.

[20] *Horizontal Burnup Gradient Datafile for PWR Assemblies*, DOE/RW-0496, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, May 1997.

[21] PARKS, C.V., DEHART, M.D., WAGNER, J.C., *Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel*, NUREG/CR-6665 (ORNL/TM-1999/303), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000.