Nuclear Criticality Safety Engineering Training
Module 16 [1]

Burnup Credit for Criticality Safety Analysis of Commercial Spent Nuclear Fuel

LESSON OBJECTIVE

This module introduces the reader to the concept of burnup credit and how to take credit for burnup in criticality safety analyses of irradiated commercial fuel. In several areas, a wide range of potential approaches are discussed. Not all the technical approaches presented have been reviewed or accepted by the U. S. Nuclear Regulatory Commission (NRC). The inclusion of a method in the discussion that follows is not intended to endorse its use or imply any level of regulatory acceptance.

The subject of burnup credit is complex and a detailed treatment of all aspects of the subject is beyond the scope of this introductory module. The references provided herein provide a source for further reading and additional applications of burnup credit.

INTRODUCTION

There are several significant differences in the complexity of criticality safety analyses for irradiated nuclear fuel compared to the analyses for un-irradiated nuclear material. Consider an un-irradiated fuel assembly. Although it is physically quite large with rather complex geometry, it is comprised of only a few materials – fuel, structural, moderator and perhaps absorber materials. The composition of each material includes perhaps a dozen nuclides, is generally uniform, and is well known (defined by the fabrication specifications). Consider this same fuel assembly after irradiation. It still is comprised of only a few materials, but the compositions of these materials now include a few thousand nuclides. Many of these nuclides are in extremely low concentrations, and many are radioactive with a wide range of half-lives. Furthermore, the power distribution, i.e., the neutron flux, varies spatially (radially and axially) throughout the reactor. Therefore, the transmutation rates of all these materials vary spatially throughout. The power distribution also varies over time depending on the reactor operating history and perhaps also on fuel management operations, making these nuclide densities both spatially- and time-dependent.

Whereas the compositions of the un-irradiated materials are well known, all of these irradiated compositions have been altered by the nuclear transmutations. It is not practical to determine these compositions with chemical measurements. Therefore, the compositions of the irradiated fuel must be predicted by depletion calculations – complex, detailed depletion calculations.

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[1] Developed for the U.S. Department of Energy Nuclear Criticality Safety Program by Dale Lancaster, NuclearConsultants.com, in conjunction with Argonne National Laboratory, Oak Ridge National Laboratory and the DOE Criticality Safety Support Group.

Furthermore, the necessary validation of the methods, models and data must address depletion as well as criticality calculations, and the greatly expanded set of nuclides in the fuel challenges the quality of both the evaluated data libraries for the calculations and the availability of integral data for the validation.

With respect to the criticality safety analyses, the most dominant reactivity effects due to the transmutations in the fuel during irradiation are the destruction of fissile atoms, the production of fission products in the fuel, production of fissile and fertile transuranic isotopes and possibly the destruction of burnable absorber atoms (if present). Three of these represent large negative reactivity effects with burnup and failure to account for these reactivity losses leads to significant conservatism in estimating the reactivity of the irradiated fuel. Production of fissile transuranic isotopes adds positive reactivity to the system, but not enough to outweigh the negative reactivity contributions of the other effects. The process of accounting for this loss of reactivity of the fuel due to irradiation, that is, taking credit in criticality analyses for the reduction in reactivity of the fuel due to burnup, is referred to as "burnup credit."

Historically, criticality safety analyses involving irradiated fuel assumed that the fuel was in its initial unburned condition. That is, this "fresh fuel" assumption did not attempt to take any credit for the burnup in the fuel. Recently, regulators have allowed credit to be taken for the reduction in reactivity of the fuel due to the burnup, and criticality safety analyses have chosen to incorporate varying levels of detail in the depletion analysis. This has resulted in several categories of burnup credit, corresponding to the level of detail in accounting for the isotopic transmutations in the fuel. These range progressively from treatment of fissile depletion only, to actinide depletion only, to actinides + limited fission products, and to all isotopes.

Burnup credit was first applied in pressurized water reactor (PWR) spent fuel pools to allow loading more fuel into a fixed size pool and to increase the enrichment limits for already manufactured fuel storage racks. It has also been utilized in spent fuel pools to compensate for the reduction in boron with degradation of BORAFLEX absorber panels. Burnup credit approval was not given for any shipping casks until 2006 although it is now utilized widely in the designs of storage casks used in the USA. Most designs are intended as dual purpose storage and transportation casks, but the approval of burnup credit in transportation is still limited to one case. Crediting burnup is extremely valuable in shipping cask design and analysis since it increases the loading capacity of a given volume cask and thereby reduces the number of shipments.

Without burnup credit, flux traps are generally used to keep the fuel array in the cask subcritical. A flux trap is a slot for water with boron absorbers on both sides. Fast neutrons are thermalized by the water in the flux trap and then absorbed or "trapped" by the boron absorbers. With burnup credit these flux traps can be removed, creating more space for fuel assemblies. A typical 24-assembly non-burnup credit cask can fit 32 assemblies using burnup credit.

SAFETY AND ECONOMIC ADVANTAGES OF USING BURNUP CREDIT

With each shipment there is a risk of a transportation accident. The higher capacity of burnup credit casks results in fewer shipments. There are large margins to a criticality event using burnup credit or non-burnup credit analysis so there is no safety difference from a criticality safety perspective. However, transportation accidents do occur; therefore fewer shipments lead to a net safety gain.

From a radiological safety point of view, use of burnup credit casks reduces personnel exposure, which mainly comes during the welding of the cask lid. Since there are fewer casks to weld shut, there is less total dose to the workers. (The dose in welding a burnup credit cask and a non-burnup credit cask is approximately the same.)

Criticality safety analysis using burnup credit is more complicated than an analysis that assumes fresh fuel. Since the isotopic content of the fuel is not directly known, it must be estimated by depletion analysis. Knowledge of the operating conditions and history at the reactor may be limited, so conservative assumptions for the depletion calculation must be established. Further, this depletion analysis requires validation. Due to the fact that there are no laboratory critical experiments with significant amounts of irradiated fuel, validation of the criticality codes and associated nuclear data libraries requires additional steps. The spatial distribution of the burnup in the fuel element is generally not known at the time of the criticality analysis, so a conservative assumption for this is also required. Finally, since the amount of burnup a fuel element has received cannot be determined by a simple visual inspection, a confirmation measurement of the burnup may be desirable or required prior to loading a shipping cask.

Although burnup credit does increase the complexity of the criticality analysis, significant margin to criticality must still be maintained. Most criticality safety analyses include an arbitrary safety margin of 5% in k (multiplication factor). The level of burnup credit permitted by regulations is still changing but in even the most aggressive forms burnup credit never challenges the established margin to criticality.

Along with potential safety advantages, burnup credit results in significant cost savings. The cost of the cask is dominated by non-basket costs. Therefore, a burnup credit cask and a non-burnup credit cask costs about the same but you need 25% fewer casks to ship the same amount of material. The costs of shipping operations are also reduced by 25%. Given the cost of casks and shipping, these savings greatly exceed the added evaluation cost due to the more complex burnup credit analysis.

Burnup credit is generally implemented with tables of required burnup as a function of initial enrichment needed to achieve a given subcritical level. These tables are often presented graphically as shown in Figure 1 and are typically called loading curves. These loading curves specify a loading criterion that limits the k_{eff} for a given application (e.g., a specific cask). Assemblies of a specified initial enrichment with burnup exceeding the burnup of the loading curve are acceptable for loading; assemblies with insufficient burnup are not acceptable for loading.

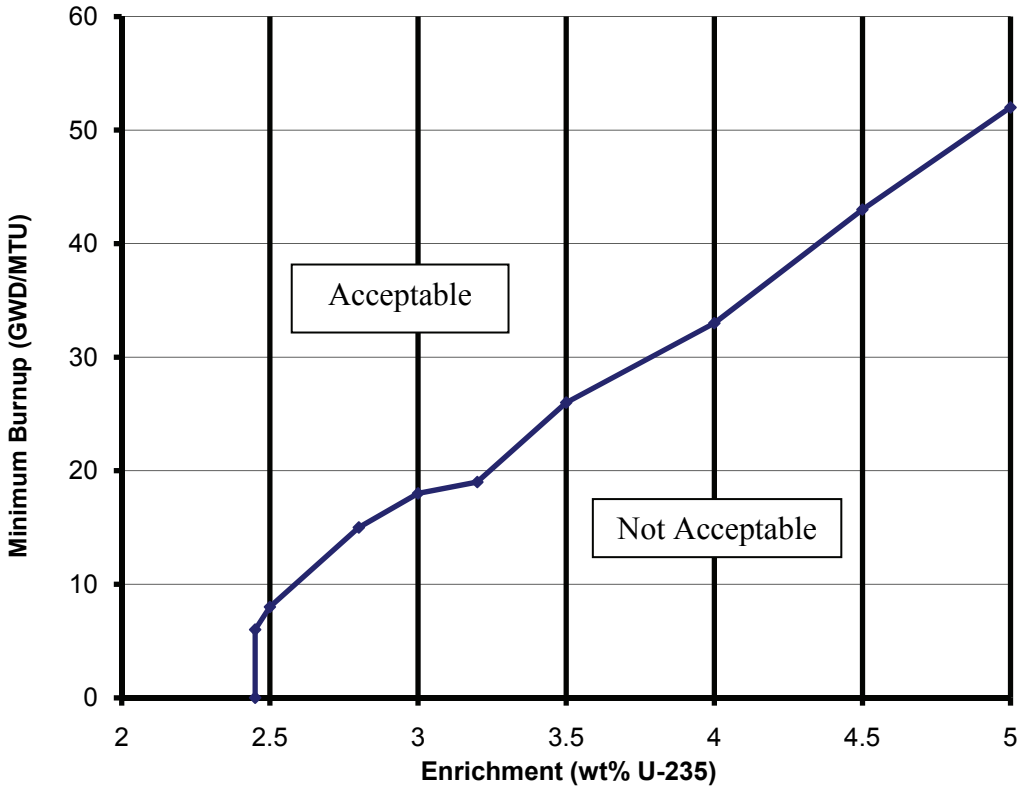

Figure 1. A Typical Loading Curve

For PWRs, fuel burnup typically decreases the reactivity of the fuel, so in those cases ignoring burnup is conservative. PWR burnup credit analyses generally do not credit the presence of fresh or relatively fresh burnable absorbers. The presence of absorbers must be accounted for if the absorber can cause an increase in discharged assembly reactivity. These issues are discussed in more detail below.

Boiling water reactor (BWR) analyses normally credit the presence of fresh Gd burnable absorbers. With increasing burnup, the Gd absorption is reduced faster than the fissile isotopes are depleted so the reactivity increases with burnup initially (see Fig. 2). At some burnup, depending on the loading of the absorbers, fuel depletion becomes the dominant reactivity change and reactivity decreases monotonically to the end of life for the bundle. A form of burnup credit is typically used for BWR plants where the Gd is credited and the peak reactivity as a function of burnup is determined. This is often called Gd credit rather than burnup credit.

REACTIVITY CHANGES WITH BURNUP

In many cases criticality safety calculations are performed for specific materials of known or specified compositions to demonstrate subcriticality, not only for normal configurations, but also for all credible off-normal conditions, such as composition, mass, density, and volume changes (e.g., double batching), addition of materials (e.g., water ingress), geometry changes (e.g., added reflection), etc.

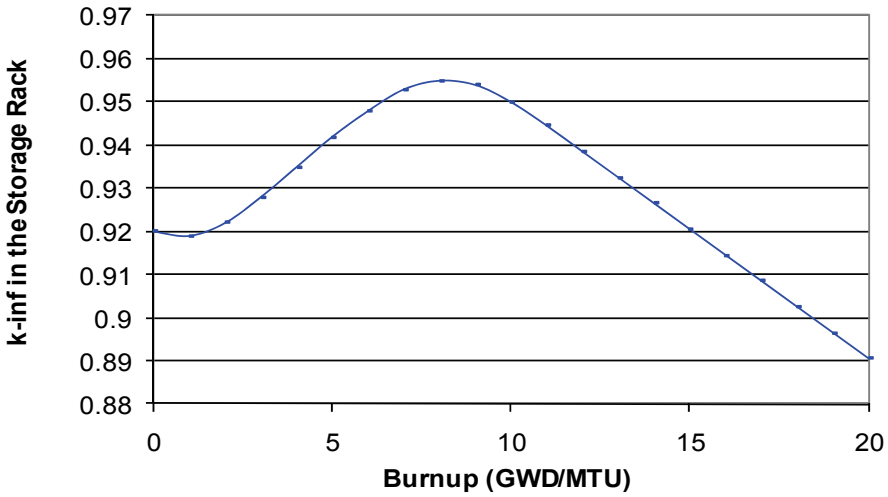

Figure 2. Example of the Change in k with Burnup for Gd-bearing Fuel

The value of burnup credit can be seen in the change in reactivity with burnup. Figure 3 shows this change for an axially and radially infinite set of 17 x 17 assemblies without burnable absorbers (in a normal core assembly pitch). As can be seen from these curves the reactivity change with burnup is roughly linear over a wide range of enrichments and burnups. The magnitude of the reactivity change as a function of burnup is strongly influenced by reactor operating conditions.

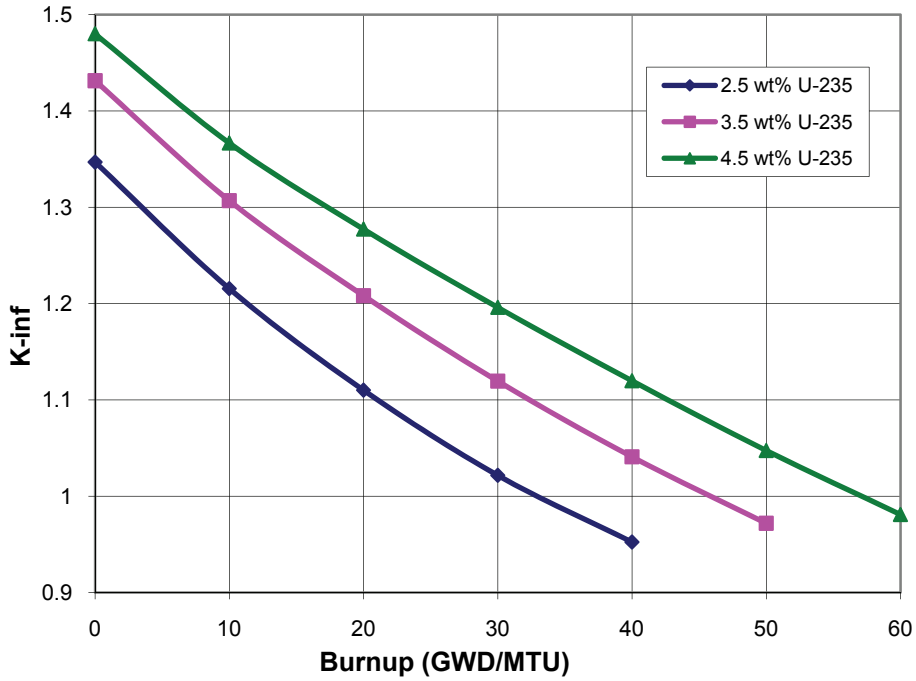

Figure 3. Change in k with Burnup and Enrichment for PWR Fuel

Burnup credit analysis often utilizes less than the full complement of isotopes in the spent fuel. Figure 4 shows the change in reactivity for four different sets of isotopes: 1) only fissile isotopes; 2) only actinide isotopes (see later section for details); 3) actinides and six fission products (Rh-103, Cs-133, Nd-143, Sm-149, Sm-152, Gd-155); and 4) all isotopes. As can be seen in Figure 4 the isotopes included in burnup credit can greatly change the calculated reactivity. Figure 4 also shows that even a limited number of fission products can result in high burnup credit return.

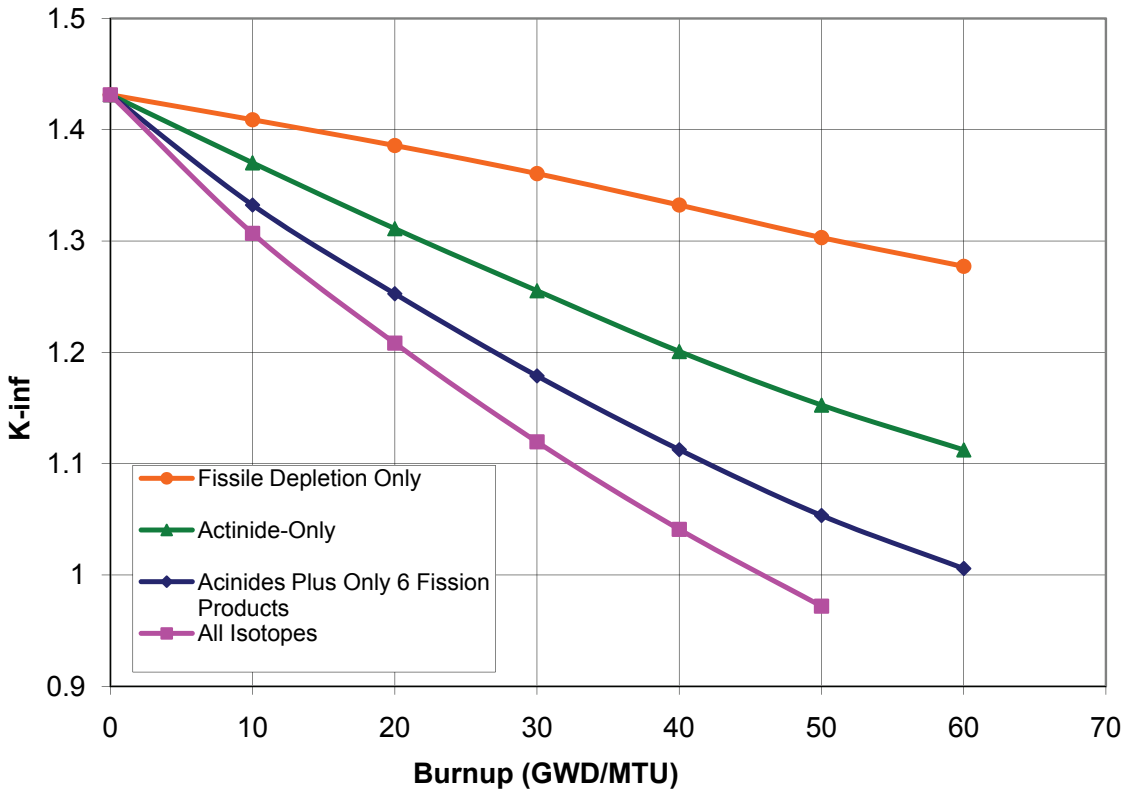

Figure 4. Change in k with Burnup and Number of Credited Isotopes for PWR Fuel

In a PWR fuel is burned at the rate of about 0.6 – 1.4 GWD/MTU per month. The fuel is in the core for about three to five years and is discharged with a burnup, in GWD/MTU, of about 10 – 12 times the initial enrichment. For example 4 wt% U-235 fuel is typically discharged with about 40 – 48 GWD/MTU burnup. (For actual discharge burnups see Figure 9).

Finally, Figures 3 and 4 represent fuel that does not contain burnable absorbers. As mentioned before, in some cases reactivity can increase with burnup due to the burnout of burnable absorbers (as shown in Figure 2 for typical BWR fuel). Notice that the reactivity peaks above the value of the initial (fresh) fuel. Gd credit analysis assumes all the fuel is at this peak reactivity regardless of the actual burnup of the fuel. Notice also that the change in reactivity with burnup becomes linear after the Gd burns out and the slope of the reactivity as a function of burnup becomes similar to that shown in Figures 3 and 4.

SENSITIVITIES OF THE REACTIVITY CHANGE WITH BURNUP

Adequate treatment of the complex transmutations occurring in the fuel during irradiation and cooling requires a thorough understanding of the sensitivities of the reactivity change with burnup. This section reviews the sensitivities to key parameters of the reactivity change as a function of burnup. Burnup credit analyses may require reactor, fuel design, and operating strategy specific sensitivity studies to demonstrate that appropriately conservative parameters are used in fuel depletion calculations.

The harder the neutron spectrum the less change in reactivity there is with burnup. Harder spectra increase U-238 fast fission and the average number of neutrons per fission for Pu-239. With a harder spectrum, for any given burnup, there is more U-235 left and more Pu-239 created than in a softer spectrum. Since burnup credit is normally correlated to burnup and enrichment without knowledge of operation history, it is important that the depletion analysis assumes the hardest spectrum possible during operation of the reactor.

The impacts of many of the parameters discussed in this section have been studied in detail for PWR plants. Many of them have not been studied in similar or sufficient detail for BWR plants since only partial burnup credit is typically used for these plants. Some important parameters, such as void and control rod history, are not discussed but are important for BWR burnup credit analysis. It should generally be assumed that all parameters discussed below relative to PWRs are also relevant and potentially important for BWR analyses even if the impacts are not discussed explicitly. In both PWR and BWR analyses, the applicability of any generic studies to each specific application should be confirmed to ensure a conservative analysis is performed.

Sensitivity to Fuel Design

Generally the higher the water to fuel ratio in the lattice the softer the spectrum will be, resulting in more change in reactivity with burnup. In general this concept is straightforward but it can cause some surprises. For example, Westinghouse reduced the fuel pin diameter to create the Optimized Fuel Assembly (OFA). Many Westinghouse plants have the original fuel, called standard fuel here, as well as OFA fuel. For a given enrichment, the extra water in the OFA assembly makes the fuel more reactive. However, with burnup the standard fuel decreases in reactivity more slowly than the OFA fuel. This means that no fuel design is most limiting for all burnups. Both fuel designs must be analyzed. Acceptable loading curves can be made using the more limiting fuel as a function of burnup or separate loading curves for each fuel type can be generated. If loading curves are generated for each fuel type, care must be taken to assure that any interface between the fuel types in the criticality safety application (pool or cask) cannot increase reactivity above the safety limits.

Generically, the impact of all fuel assembly designs present at the plant need to be considered in the analysis. This could include different manufacturing tolerances if fuel has been supplied by multiple vendors. An understanding of the differences of the fuel assembly designs used over the life of the plant is important. The effort needed to identify the limiting design or designs can be significantly reduced if the differences among assembly designs are small and/or in neglected components.

Sensitivity to Enrichment

The decrease in reactivity with burnup is not very sensitive to enrichment for low enriched fuel (less than 5 wt% U-235). A higher enrichment hardens the spectrum slightly but the spectrum is dominated by the moderation rather than the fissile absorption. As seen in Figure 3, the slope of the curves is similar for the range of enrichments important to PWRs. Finally note that burnup credit is normally presented as a required burnup as a function of initial enrichment. With this approach any sensitivity to enrichment is implicitly included in the calculations.

Sensitivity to Fuel Temperature

Higher fuel temperatures increase plutonium production through Doppler broadening as well as spectrum hardening. The Doppler reactivity effects are dominated by the U-238 capture resonance. The sensitivity of the final system reactivity to the fuel temperature used in the depletion analysis may be on the order of 4-5 pcm/K [1].

The fuel pellet temperatures depend on the linear heating rate, pellet thermal conductivity, and the surface temperature. The surface temperature depends on the pellet-clad gap size, composition, and cladding corrosion and crud levels. The fuel temperature initially decreases as the pellet-clad gap closes then increases with burnup as corrosion increases and pellet thermal conductivity decreases. Fuel temperatures during irradiation cannot be measured. Fuel management computational tools produce temperatures that are burnup dependent. Normally the fuel temperature is not an input parameter for fuel management codes since it is calculated to be consistent with the power density.

A typical assumption for the fuel temperature in a PWR is about 1000 K[1]. More detailed analysis can be performed to establish an appropriate fuel temperature for the specific application being considered.

Sensitivity to Moderator Temperature and Density

Higher moderator (water) temperature and the accompanying decrease in moderator density harden the spectrum and produce more plutonium. The sensitivity of the final system reactivity to the water temperature (density) in the depletion analysis can be as high as 35-90 pcm/K [1]. The moderator temperature increases from the bottom to the top of the core. It is conservative therefore to use the outlet temperature for the depletion analysis. Due to radial peaking the average core outlet temperature is non-conservative for some assemblies so a burnup averaged radial peaking factor may be needed in determining the appropriate outlet temperature. The outlet temperature depends on the plant and peaking factor but a burnup-averaged hot assembly outlet temperature for PWRs can be around 610 K (moderator density of 0.628 g/cm^3). Plant specific confirmation of an appropriate maximum assembly outlet temperature should be performed. Some analyses assume different axial temperature distributions. This may be of little value since the top of the fuel dominates reactivity due to the end effects at high burnups.

Sensitivity to Soluble Boron Concentration in PWR Operations

Soluble boron is used to control the normal change in reactivity with burnup for a PWR. A higher soluble boron concentration leads to a harder spectrum and therefore more plutonium production. Using an average concentration during depletion is generally an acceptable approximation for calculating the reactivity of the fuel even though the concentration generally starts high each cycle and decreases to near zero at end of each cycle. The sensitivity of discharged assembly reactivity to the soluble boron concentration used in the final application during the depletion may be as high as 3 to 3.5 pcm/ppm. [1]

The maximum soluble boron concentration in a PWR is limited by moderator temperature coefficient limits, primary chemistry considerations, or other operational constraints. Higher soluble boron concentrations produce more positive moderator temperature coefficients. The soluble boron concentration used during depletion calculations should be selected to exceed the maximum burnup averaged concentration. Current, past, and future fuel management should be considered to ensure that the concentration used provides adequate conservatism for all fuel covered by the analysis.

Sensitivity to Burnable Absorber and Control Rod Modeling in PWR Depletion Analysis

Burnable absorbers (BAs) are materials with a high thermal absorption cross section that are added to fuel to control the reactivity during depletion. Burnable absorbers have also been referred to as burnable poisons (BPs). The burnable absorbers are used to lower soluble boron concentrations and therefore meet moderator temperature coefficient constraints. They are also used to flatten the power distribution. BAs can be absorber material added to the fuel rod pellets or rods containing boron inserted into the guide tubes of fuel assemblies not under control rods. The rods inserted into the guide tubes are usually removed after the first cycle of operation and therefore are called removable burnable absorbers. (In some cases these removable burnable absorbers are used in the second cycle of operation.) The burnable absorbers that are part of the fuel pellets are called Integral BAs (IBAs). The most common absorbers are Gd_2O_3 or Er_2O_3 mixed into the fuel pellets or a ZrB_2 coating on the outside of the fuel pellets.

Control rods in most PWR plants consist of a number of individual rodlets mounted on the same axially translating hub. The number of rodlets varies depending on the fuel assembly array, but is typically between 16 and 24. Many different materials are used in control rods; boron carbide (B_4C) and silver-indium-cadmium (AIC) are probably the most common. During power operations, most of the control rods are fully removed from the core, though a small number of rods may be slightly inserted at the top end of the fuel assemblies. The insertion of control rods causes spectral hardening because of both large thermal absorption cross sections and also water displacement from the control rod guide tubes. Both these effects increase the reactivity of discharged fuel assemblies.

Burnable absorbers and control rods harden the spectrum and increase plutonium production. Ignoring these in the analysis would be non-conservative. However, the effect of burnable absorbers and control rods is large enough that care should be taken to match the actual effect of these materials.

The depletion analysis could assume the maximum number of BA pins per assembly and the design that displaces the most water in the fuel assembly. This would be conservative but would result in a large burnup credit penalty. Alternatively, the depletion analysis could consider the fuel management of the plant being analyzed and bound the BA use and strategies for past, current, and future design strategies. In this context, a larger number of BAs inserted for a longer burnup would be bounding as it would maximize the integral spectral hardening effect.

Gd and Er integral burnable absorbers may be ignored in PWR analysis given proper and sufficient justification. There is a positive reactivity effect due to the spectrum hardening but both of these burnable absorbers produce isotopes that have a negative reactivity effect. This residual poison effect can be greater than the positive spectrum hardening effect.

ZrB_2 coated pellets harden the spectrum and the residual absorption is less than the spectrum hardening effect. The positive reactivity effect is typically small, probably on the order of 0.5% Δk_{eff} or less.

Oak Ridge National Laboratory (ORNL) has written two reports on burnable absorbers and burnup credit [2, 3]. Figure 5 is taken from Reference 3 and shows the magnitude of the effect of removable BAs on burnup credit.

Control rods inserted in the core during depletion have an even larger effect on the depletion analysis than BAs. However, control rods are generally out during operation and therefore have no effect. If operation with inserted controls rods is done then a separate loading curve for those assemblies that were subject to those operation conditions may be desirable due to the small number of assemblies that would be affected by rods. Depending on the magnitude of the control rod penalty, it may be preferable to generate a single loading curve including this penalty to simplify implantation of the analysis. ORNL has published a study on control rod effects [4].

Some PWRs have included guide tube inserts for various purposes, including limiting the neutron fluence to the pressure vessel. These inserts harden the spectrum and need consideration in the depletion analysis. In some cases the devices are part-length axially and may not affect the flux at the axial heights of concern, but may yield an atypical axial burnup distribution. The associated hardware to position any part-length absorbers should also be considered if it causes spectral hardening via water displacement from the guide tubes. Careful review of the usage of these devices or any other guide tube inserts that may be used in a small percentage of the fuel is needed so that the burnup credit is conservative without overly penalizing the rest of the fuel.

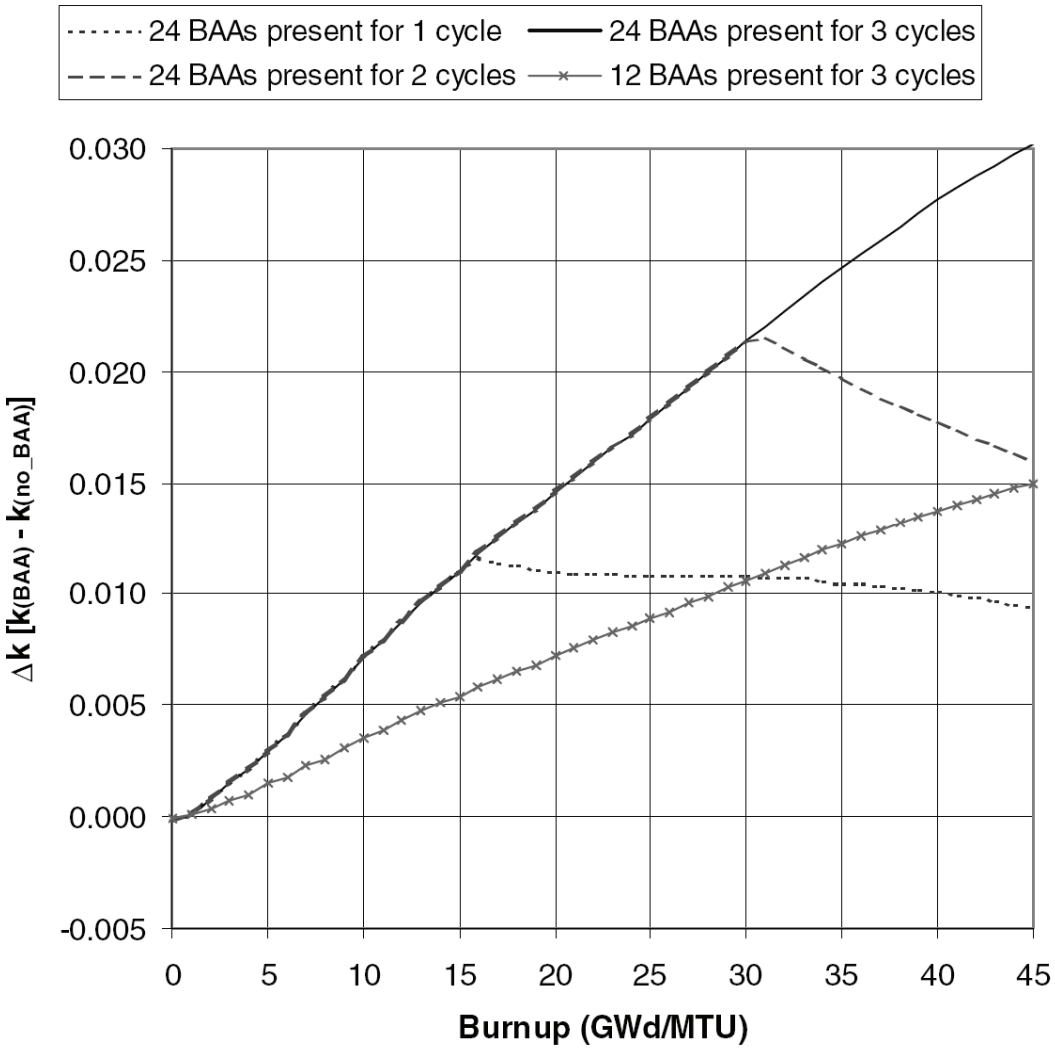

Figure 5. Reactivity Effect of BA Removal Assumptions [3]

Sensitivity to Operating History and Specific Power

Depletion analyses need to consider the way the power plant was operated in order to correctly predict the isotopic content and resulting burnup credit.

Higher specific power (MW/MTU) increases the equilibrium Xe content in the fuel, thereby hardening the spectrum and producing more plutonium. This spectral effect is very small. There is another effect related to Gd-155. Gd-155 comes from the decay of Eu-155 which has a 5 year half-life. With lower specific power, more of the Eu-155 decays to Gd-155 during operation. This Gd-155 is then lost through absorption during power operation. The loss of the Gd-155 makes the spent fuel more reactive in the final criticality safety application (storage rack). The specific power effect is sufficiently small so that for an analysis which includes Gd-155 it is appropriate to use the nominal specific power. For actinide-only analysis a high specific power is used which has very little effect on the final reactivity. A typical specific power for actinide-only analysis is 60 MW/MTU.

Power plants have shutdown periods between cycles and may not operate at full power all the time. This could have an effect on burnup credit analysis. ORNL studied this effect [5] and found that it was rather small. The conclusion of the operating history study is that it is appropriate to use a continuous operation cycle with no downtime, provided margin is present for operating history uncertainty.

Sensitivity to Cooling Time

After shutdown the reactivity of the fuel decreases. In the initial hours (72 -100 hours) after shutdown there actually is an increase in reactivity but most plants are precluded from handling fuel during this time because of the decay heat. If no technical specification exists to determine the minimum cooling time, the time of the reactivity peak in this range should be used.

Burnup credit applications often use cooling time (time after reactor shutdown) to gain more burnup credit. Figure 6 shows reactivity as a function of time [6]. (Note that Figure 6 suggests that there is a decrease in reactivity for the Actinide-Only curve, due in part to Eu-155 decaying to Gd-155. In Actinide-Only analysis there is no Eu-155 or Gd-155 accounting.) As can be seen in Figure 6 there is an increase in reactivity with long cooling times. Assuming a cooling time of no greater than 20 years is credited, the analysis is conservative for the first 1000 years, which is longer than the expected life of a cask or rack outside of repository applications.

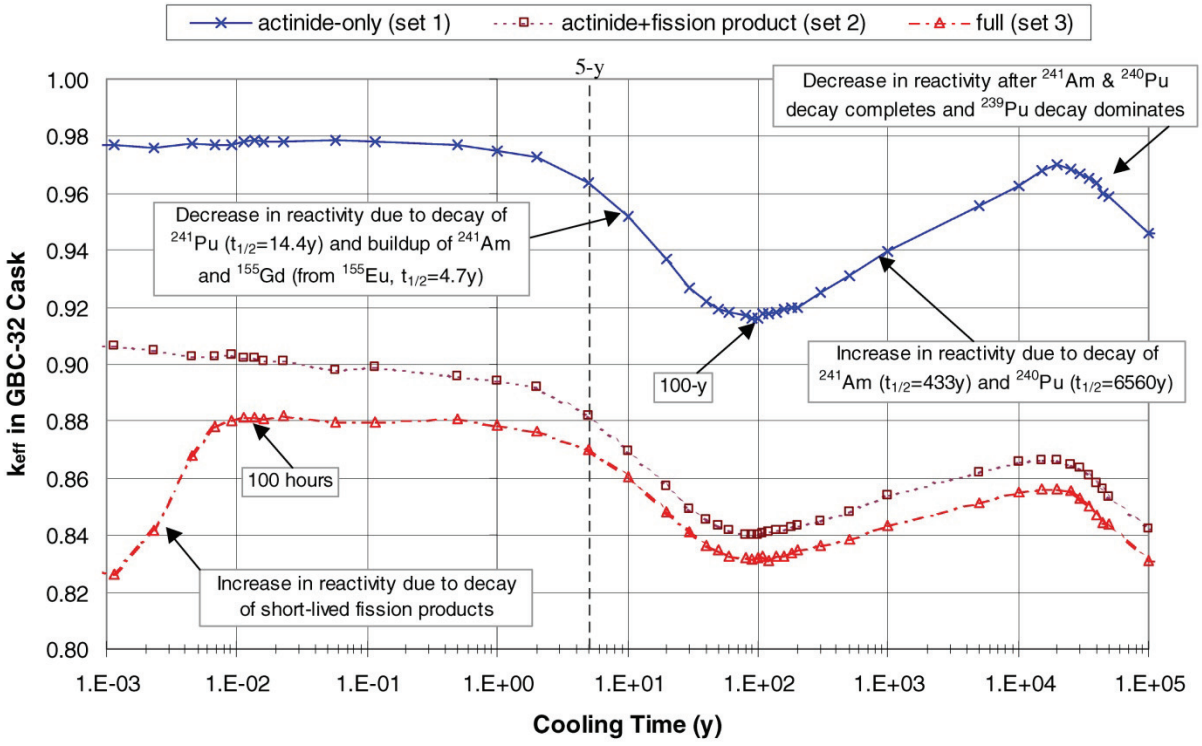

Figure 6. Change in Reactivity with Cooling Time [6]

BURNUP CREDIT CATEGORIES

Burnup credit is typically categorized by the isotopes that are used in the final cask or rack criticality safety calculations. The depletion calculations should always use all relevant isotopes. Full burnup credit utilizes an extended set of actinides and fission products, limited primarily by consideration of nuclide mobility and radioactive decay. Actinide-only burnup credit only uses the actinides found in UO_2 or MOX critical experiments. Fissile depletion burnup credit accounts for changes only in the fissile isotopes (U-235, Pu-239, and Pu-241). A fourth category of burnup credit uses actinides and selected fission products. The change in reactivity for these four types of burnup credit was shown in Figure 4.

The selection of a particular set of isotopes for use in the burnup credit analysis depends on a wide range of technical and regulatory factors. The single most important consideration should be the worth of a particular isotope. The importance of each considered isotope to the overall reactivity of the system is important because of the effort needed to generate and validate the number densities used in the analysis models. Isotopes with long half-lives, including stable isotopes, are good candidates to include as there is little question about their presence as a function of cooling time. This is especially true for isotopes that contribute negative reactivity to the system. It is desired that sufficient data are available to provide a thorough validation of the production of any isotope to be credited within the code system used for fuel depletion calculations. This availability of data for validation can be a key consideration. In some cases, volatile species such as cesium or noble gases are neglected because they may escape from the fuel during storage. Finally, any regulatory documents that provide guidance on acceptable nuclides or acceptable methods for selecting nuclides should also be considered when selecting which burnup credit category and which specific isotopes to credit.

Fissile Depletion Burnup Credit

Fissile depletion burnup credit was suggested as a conservative approach for early burnup credit. By now, this level of conservatism is not needed for PWR fuel but may be useful for non-power plant fuel. Due to safeguard requirements, the U-235, Pu-239, and Pu-241 content may be well established. Validation data for the cross sections of these isotopes are extensive. For fuel where the operational data are less routine, a conservative selection of isotopes may be appropriate; however, a conservative depletion analysis will still be needed.

Note that Am-242m is not included here (or in Actinide-Only) and that it is a fissile isotope. The concentration of Am-242m is small even at large burnups, so the impact on k is close to the normal uncertainty in the analysis. Furthermore, Am-243 is also not included, and it has a negative worth that is larger than the positive worth of Am-242m. Finally Cm-242 and Cm-244 are ignored since their concentrations are too small to have a noticeable effect on k.

Actinide-Only Burnup Credit

In actinide-only burnup credit the changes in concentrations of nine isotopes are considered: U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241 [7]. The concentrations of these nine isotopes are generally determined in any chemical assay of spent

fuel, and their reactivity worths have been demonstrated in critical experiments. U-236 and Np-237 are two important actinides that are generally not included in actinide-only burnup credit since they are not found in many critical experiments.

Figure 7 shows the fraction of the total absorption for the actinides based on 4.5 wt% U-235 at 50 GWD/MTU [1].

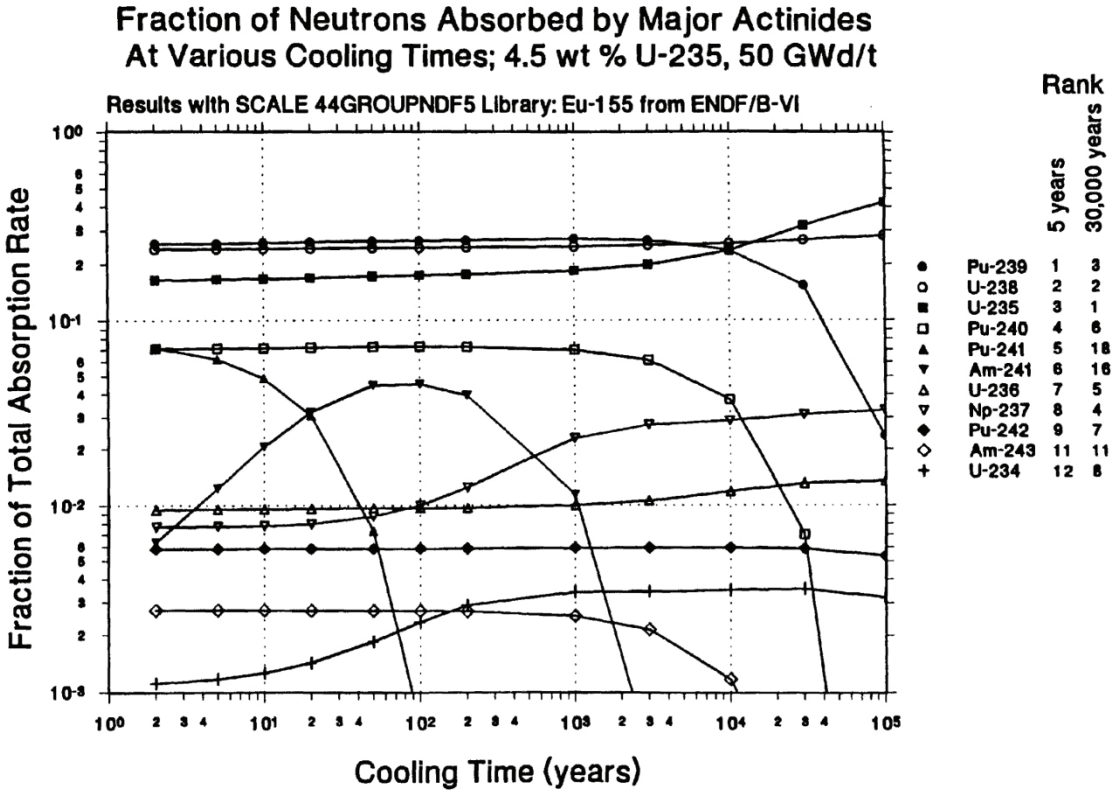

Figure 7. Relative Absorption of Actinide Isotopes [1]

Actinides Plus Fission Product Burnup Credit

As can be seen in Figure 4 the inclusion of fission products significantly increases the burnup credit. Unfortunately, fission products are not normally included in the criticality benchmark experiments and the number of chemical assays that measure fission products is limited. Figure 8 shows a comparison of the absorption rates of the top 20 fission products. A subset of the top six fission products (Nd-143, Sm-149, Rh-103, Sm-152, Cs-133, and Gd-155) has been selected by the French for their burnup credit applications. For these isotopes there are sufficient chemical assays and the French have done critical experiments with these same isotopes. The method of conservative validation is still under review.

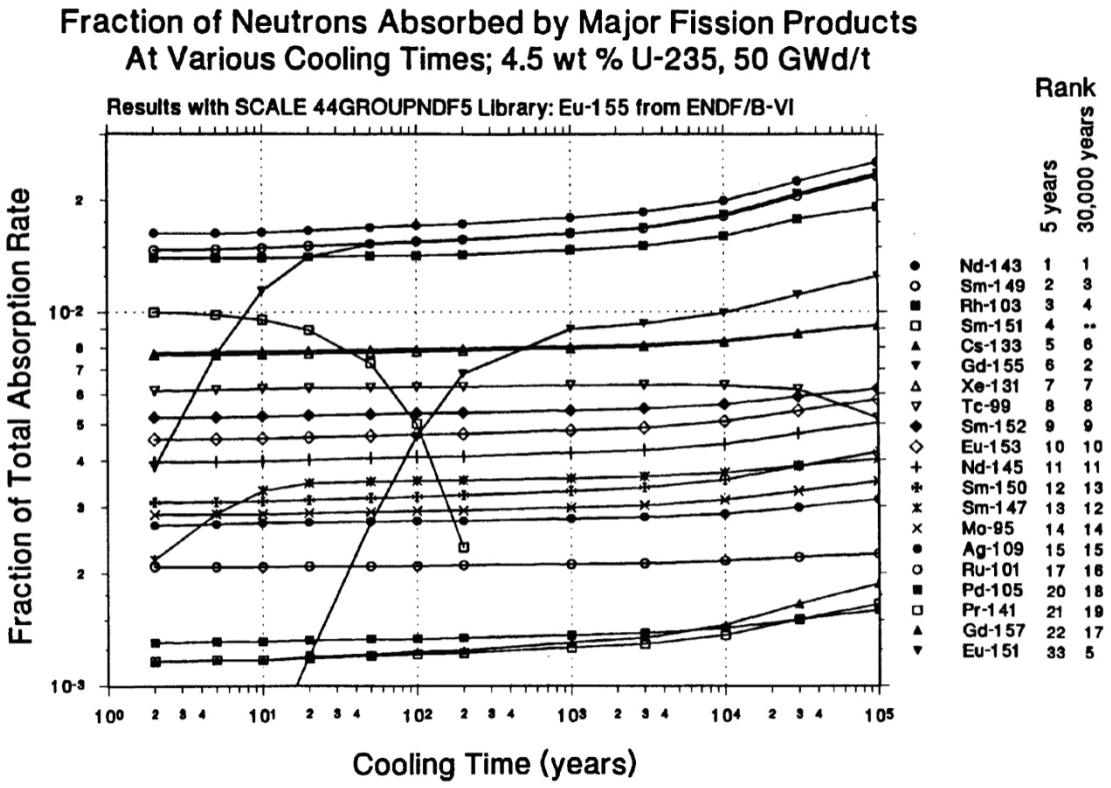

Figure 8. Relative Absorption of Fission Product Isotopes [1]

Full Burnup Credit

Full burnup credit utilizes all isotopes. In order to utilize full burnup credit a bias and uncertainty in the determination of the reactivity change due to depletion must be established. It may be possible to determine the bias and uncertainty from power reactor measurements. No complete methodology for doing this has been presented to or accepted by the NRC or the Department of Energy (DOE). Full burnup credit has historically used engineering judgment of the depletion uncertainty of this reactivity change and therefore has only been used in situations where there is a large safety margin and defense in depth. For example, PWR spent fuel pools do not credit most of the soluble boron in the criticality analysis so full burnup credit has been acceptable. Efforts to replace the engineering judgment with rigorous documentation may allow full burnup credit for more applications.

VALIDATION

For burnup credit both the isotopic content and the reactivity worth of the isotopes need to be validated. This can be done by chemical assays and criticality experiments or by validating the change in reactivity due to burnup.

Validating Isotopic Content

The computer codes used for depletion analysis can be validated by use of chemical assays of spent fuel. Taking a chemical assay of spent fuel is expensive so there are fairly limited data. The OECD/NEA maintains a website to provide data on publicly available chemical assays and calls the database SFCOMPO. SFCOMPO gives sufficient details to model each chemical assay. The website is http://www.nea.fr/sfcompo/Ver.2/Eng/index.html. For chemical assays that have not yet been distilled into easy to use data, there is another website, http://www.nea.fr/science/wpncs/ADSNF/reports/. Table 1 summarizes the available data. From Table 1 it can be seen that there are only 32 chemical assays from PWR fuel that are high quality and contain fission products. Half come from one Japanese reactor. The lack of a large data set for validation of fission product content has caused delays in implementing burnup credit utilizing fission products in transportation safety analysis. Table 1 also demonstrates that there is a large data set for validation of actinides in spent fuel. Figure 9 shows how the currently available chemical assays compare to the discharged fuel in burnup and enrichment.

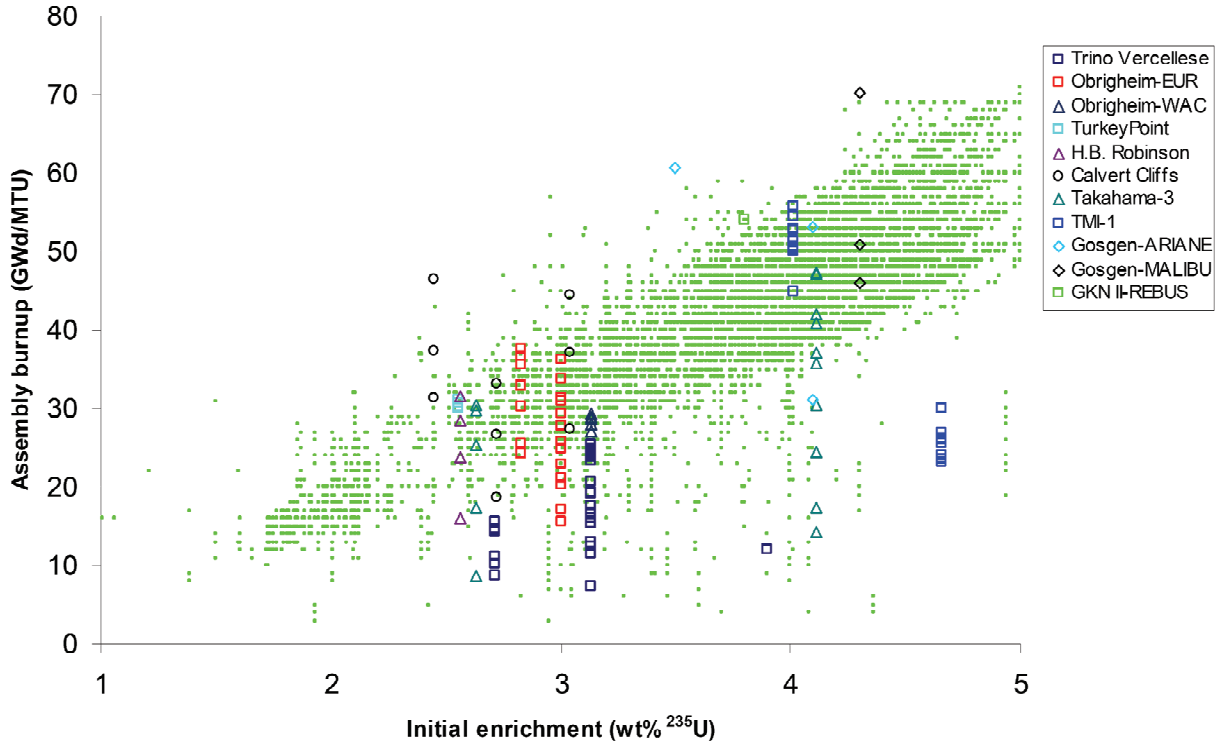

Figure 9. The Distribution of Chemical Assay Data Compared to the Range of Discharged Fuel (shown as green dots) [8]

Table 1. Chemical Assay Data				
Reactor	**Reference**	**Total Number of Assays**	**Number of Assays with Fission Products**	**Comments**
Obrigheim	SFCOMPO	23	0	
Obrigheim	DOE/RW-0497	6	0	Half assembly
Gundremmingen	SFCOMPO	12	0	BWR
JPDR	SFCOMPO	30	0	BWR
Tsuruga-1	SFCOMPO	10	0	BWR
Fukushima-Daini-2	SFCOMPO	18	18	BWR
Mihama-3	SFCOMPO	9	0	
Genkai-1	SFCOMPO	2	0	
Takahama-3	SFCOMPO	16	16	
Cooper	SFCOMPO	6	0	BWR
Monticello	SFCOMPO	30	0	BWR
Calvert Cliffs	SFCOMPO	9	3	
H. B. Robinson	SFCOMPO	6	0	
Trino Vercellese	DOE/RW-0497	14	0	Unusual lattice PWR
Turkey Point	DOE/RW-0497	5	0	
Yankee Rowe	DOE/RW-0497	8	0	Unusual lattice PWR
Vandellos	ENUSA-CSN	9	9	
TMI	NUREG/CR-6968 [9]	19	19	Poor Quality
Gosgen	NUREG/CR-6969 [10]	6	6	3 Proprietary
GKN II	NUREG/CR-6969 [10]	1	1	

The chemical assays need to be analyzed with the same code, nuclear data, and options that are used for the criticality application (rack or cask). The reactivity worth of the deviation between predicted and measured isotopic content is used to determine a bias and uncertainty in the criticality application. This is done by analyzing the rack or cask with two sets of atom densities for each chemical assay: the predicted and measured atom densities. The differences in k in the cask/rack for each assay are analyzed to determine the mean difference and the uncertainty in this mean. By this method a bias and uncertainty for the k of the cask/rack due to the depletion analysis is determined. For further details, see the "direct difference method" in Reference 8. This is possible for actinide-only burnup credit since there is a large set of chemical assays that contain all the actinides credited in the final cask/rack analysis.

For burnup credit that utilizes fission products, the atom densities of the fission products can be adjusted or "corrected" to conservatively match the chemical assays. The "correction" approach is needed since there are not enough chemical assays that contain all the isotopes used in the final cask/rack analysis. The bias and uncertainty approach described for actinide-only burnup credit can still be used for the actinides so only the fission products are adjusted in this method. This correction approach means each isotope is individually adjusted in the conservative direction. This one-direction correction for each isotope grossly under predicts the absorption of the fission products but a statistical combination of the corrections is difficult since the reactivity worth per atom of the isotopes is different. The "correction factors" for each isotope are calculated by finding the mean bias in the atom density for the set of chemical assays and then determining the correction factor that is used to multiply the calculated atom densities to assure covering the mean bias and its uncertainty. For more details refer to References 7 and 11. Other approaches may be developed in the future that will be less conservative.

Validating Criticality Calculations for Burnup Credit

As for all criticality safety analyses, validation of the criticality code, cross sections and methodology is required. In the case of burnup credit there is no benchmark experiment that contains actinide and fission product compositions consistent with spent fuel. Therefore, burnup credit requires a unique approach to validation.

The French have performed and documented a set of experiments known as the HTC experiments [12] that are particularly useful for validation of the primary actinides. These experiments use pins that contain uranium and plutonium compositions consistent with spent nuclear fuel of 4.5 wt% U-235 burned to 37.5 GWD/MTU. These experiments are very similar to burned fuel in a burnup credit application, and are of correspondingly high value in burnup credit validation. They should be included, with the caveats noted in Reference 12, for burnup credit validation. Unfortunately, these experiments do not contain fission products.

Burnup credit criticality validation should be performed using the HTC experiments, supplemented with available MOX critical experiments documented in the International Handbook of Evaluated Criticality Safety Benchmark Experiments [13]. The HTC experiments are needed because none of the publicly available mixed uranium and plutonium critical experiments include low enrichment uranium and plutonium nuclides in compositions similar to burned fuel compositions. The HTC experiments reports are available from Oak Ridge National Laboratory for use consistent with a non-disclosure agreement that must be signed prior to receiving the data.

For actinide-only burnup credit, it is also possible to validate the criticality safety analysis for the actinides by use of fresh UO_2 critical experiments in conjunction with MOX critical experiments. The bias and uncertainty (or Upper Subcritical Limit - USL) may be determined separately for the MOX set of experiments and the UO_2 set of experiments. The most limiting bias and uncertainty (or USL) are assumed for the criticality safety analysis of the application.

It would be more appropriate to pool the fresh UO_2 and MOX experiments and evaluate trends, including one for plutonium content {e.g., g Pu/(g Pu + g U)}. This may provide a slightly less conservative bias and bias uncertainty.

Figure 10 displays sample validation results, as a function of the average neutron energy causing fission. This is a spectral measure that is typically used to investigate the validation experiments for trends in bias and bias uncertainty. This trending analysis is similar to many other criticality validation approaches, though trends should also be investigated for plutonium content, as mentioned previously. As with other criticality validations, the burnup credit validation must be performed for the same code and cross section library that will be used in the application. It should be expected that the introduction of a new cross section library, for example one based on ENDF/B-VII, will yield different results from older libraries, such as those based on ENDF/B-V.

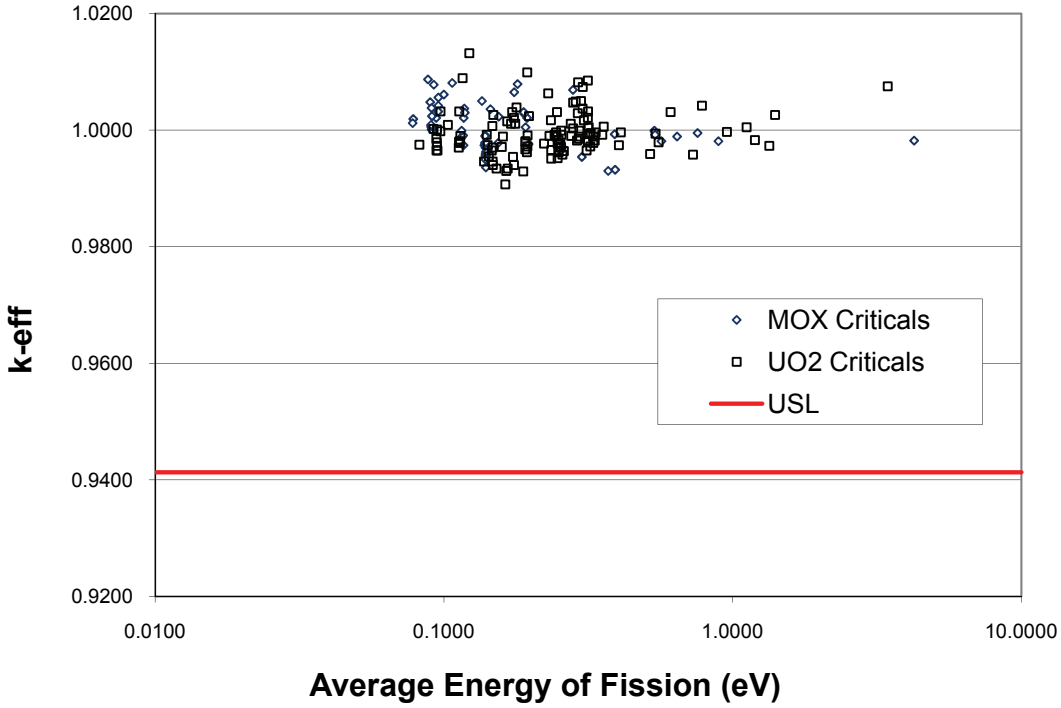

Figure 10. Margin Between the Critical Experiments and the Upper Subcritical Limit (USL)

For burnup credit that includes fission products, some validation of the worth of the fission products is required. This can be accomplished for some fission products using fission product worth experiments. With a range of experiments that cover the neutron energy spectrum for the final application (cask/rack), it is possible to estimate a correction factor for the presence of some fission products. Critical reactor data containing fission products may be used to supplement the validation. Use of the reactor data gives confidence that none of the individual fission products are in error by enough to invalidate the analysis. However, additional margin is required since the critical reactor data will not allow for determination of the quality of individual fission product cross sections and cancellation of errors could hide some problems. Methods are

currently under development by the Electric Power Research Institute (EPRI) and others to support use of critical reactor data to validate full burnup credit. Although burnup credit using some fission products has been licensed in the USA for a cask, the method used is proprietary. The use of fission products in burnup credit has not yet been implemented in Europe.

Validation Using Measured Reactivity Change Due to Depletion

Prediction of the reactivity state of power reactors is a requirement of operation. Based on experience with the accuracy of fuel management code reactivity predictions, an engineering approximation that 5% of the reactivity change due to depletion can be used for the depletion uncertainty has been used in spent fuel pool burnup credit analyses in the United States. Research is ongoing to convert this engineering judgment to a well-documented methodology. Use of the so called "5% of the reactivity decrement method" is still being accepted for full burnup credit in spent fuel pools. Continued use of this depletion uncertainty may be impacted by ongoing research.

SPATIAL DISTRIBUTION OF BURNUP (END EFFECTS)

In many criticality safety analyses the spatial distribution of the important isotopes is well known. However, in burnup credit analysis the spatial distribution of the important isotopes depends on the operation history of the fuel in the reactor. The finite nature of the reactor causes a near cosine distribution of the axial flux and burnup in the initial core, but the variation in moderator temperature shifts the power distribution towards the bottom of the core. Burnup flattens this power distribution and by discharge the burnup distribution is approximately flat with lower burnup near the ends. Generally, while both ends of the assembly have reduced burnup compared to the central portion of the fuel, the burnup at the top of the fuel is lower due to the higher operating temperatures. As burnup increases beyond about 30 GWD/MTU, the power distribution shifts towards the relatively fresh fuel at the ends of the assembly. This generally causes the axial gradients to flatten somewhat at higher burnups. Radially, tilts in the burnup across each assembly also occur, but due to fuel shuffling these radial tilts may flatten somewhat before the fuel is discharged. This section addresses the axial and horizontal distribution of burnup levels needed in burnup credit analysis.

Limiting Axial Burnup Profile

Neutron leakage at the axial edges of the core reduces the local power and hence the burnup. Due to the moderator temperature increasing as it goes up the core, the top of the fuel has the least burnup in the assembly. In early burnup credit analyses, the burnup was assumed to be axially uniform. It was discovered that at high burnups, modeling the axial burnup profile produced a higher reactivity. The term "end effect" was defined as the difference in reactivity of a system modeled with an axial burnup profile minus the reactivity of the same system modeled with uniform axially-averaged burnup. With typical axial burnup profiles, the "end effect" turns positive between about 20 and 30 GWD/MTU. With worse shape profiles, the "end effect" can turn positive at burnup levels as low as 10 GWD/MTU. The burnup value of this transition from a limiting uniform burnup profile to a limiting distributed burnup profile is also a function of enrichment, and must be treated explicitly for all burnup credit analyses. While it is not

necessarily important to determine when the limiting profile changes, it is important to know that a limiting profile is used at all burnups considered in the analysis. In some complex storage configurations that include fresh fuel, the transition to a limiting distributed burnup profile may occur at higher burnups than expected due to neutronic communication with fresh assemblies.

DOE collected a set of 3169 end-of-cycle calculated axial burnup profiles from the utilities with PWRs [14]. Parish and Chen calculated the "end effect" for each shape [15]. Figure 11 shows the results of this analysis. The shapes were then grouped into burnup bins and ordered in each bin by the magnitude of the end effect. Kang and Lancaster confirmed that this ordering was appropriate for casks and reduced the number of bins to three: 0-18 GWD/MTU, 18-30 GWD/MTU, and ≥ 30 GWD/MTU [16]. The three shapes selected from the DOE database include a shape that came from rodded operation and a shape that came from a transition to an axially blanketed core. These selected shapes are considered robust enough to cover all full-length fuel that (1) does not include any axial enrichment zoning, (2) has no significant depletion with control rods present, (3) is enriched to 4 wt% U-235 or less, and (4) is similar to a currently operating commercial PWR. The selection process was reviewed by ORNL [17] which supported the conclusion on robustness of the shapes and confirmed that the three selected shapes are limiting. The ORNL report provides limiting axial burnup profiles for each of 12 assembly average burnup ranges. Figure 12 shows the selected shapes and the average shape for the same burnup range.

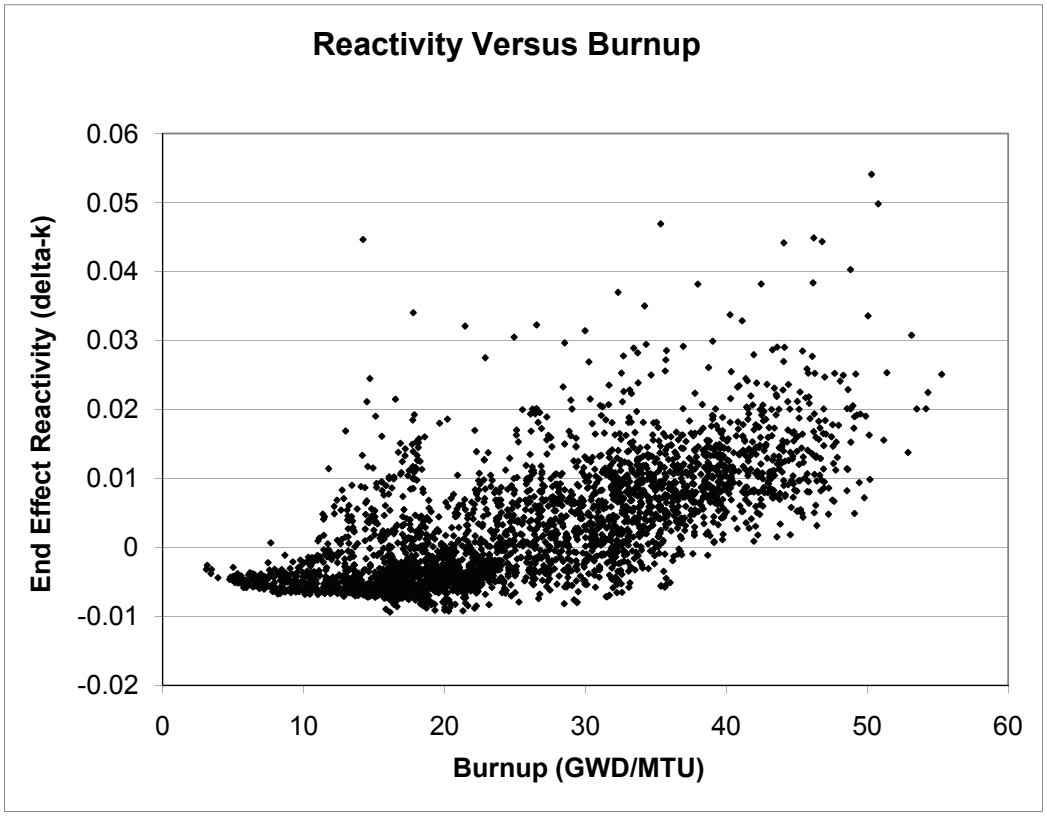

Figure 11. End Effects of DOE Axial Burnup Profile Database [15]

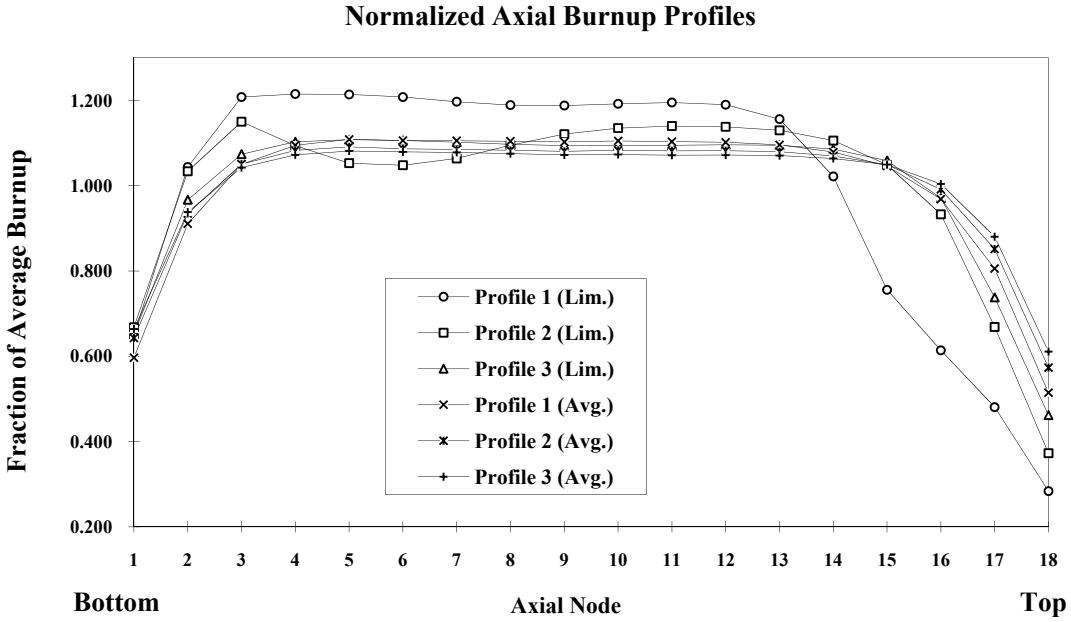

Figure 12. Selected Limiting Axial Burnup Profiles Compared To the Average Axial Burnup Profile For Each Burnup Bin [16]

The limiting axial profiles are valid for uniform enrichment. If axial blankets are used in the fuel, more analyses may be required to determine appropriate profiles for use. In some cases, it may be conservative to ignore the blankets and use the enrichment of the non-blanket fuel and the limiting non-blanket shape. The presence of blanketed fuel assemblies next to unblanketed assemblies in the core is likely to generate a more limiting axial burnup profile in adjacent unblanketed fuel. The end node burnups will be suppressed by the low reactivity blankets that are located in adjacent blanketed assemblies. These profiles may be the most limiting axial profiles, and need to be considered in addition to axial profiles from completely unblanketed core designs. It is desirable to gain the reactivity benefit of the axial blanket but for this benefit new limiting shapes must be determined.

If the fuel does not meet the requirements of the DOE database or benefit from specific plant data is desired, plant specific burnup profiles can be used. If plant specific data are used care must be taken to assure that all fuel needing burnup credit is covered by these shapes. It may also be desirable to institute a screening process to ensure that future shapes fall within the envelope covered by the analysis.

Limiting Horizontal Burnup Profile

The fuel assemblies also have a radial burnup profile. If the gradients are small, the k of the system may not be dependent on the pin-by-pin burnup but rather the burnup variation over a larger volume. It may be appropriate to model the radial burnup gradient as one-half of the assembly with a high burnup and the other half with a low burnup. In some cases, it may be necessary to model different burnups in each quadrant of an assembly. The implementation of this modeling in the analysis may result in modeling isotopic number densities for each pin if

automation can be developed to implement this level of detail efficiently. The implementation of radial tilts within the depletion code should be done with care, and may be difficult or cumbersome. The criticality model of the rack or cask will also need to be reviewed to ensure that limiting configurations considering stored assembly rotations have been identified or controlled.

The database for determining an appropriate radial burnup profile is very limited [18]. It comes from several cycles of one plant. Due to the limited database the treatment of the data is very conservative. The database gives the burnup by quadrant and the average burnup for each assembly. The ratio of the maximum quadrant burnup to the assembly average burnup was plotted and is reproduced here as Figure 13. Using Figure 13, for assemblies with less than 18 GWD/MTU burnup, one-half of the assembly is modeled as 33% less than the average burnup and the other half is modeled as 33% more than the average burnup. Similarly in the range between 18 and 30 GWD/MTU burnup each side is lowered or raised by 25%. Finally, for assemblies with greater than 30 GWD/MTU burnup a 20% increase and decrease is used to establish the limiting burnup profile.

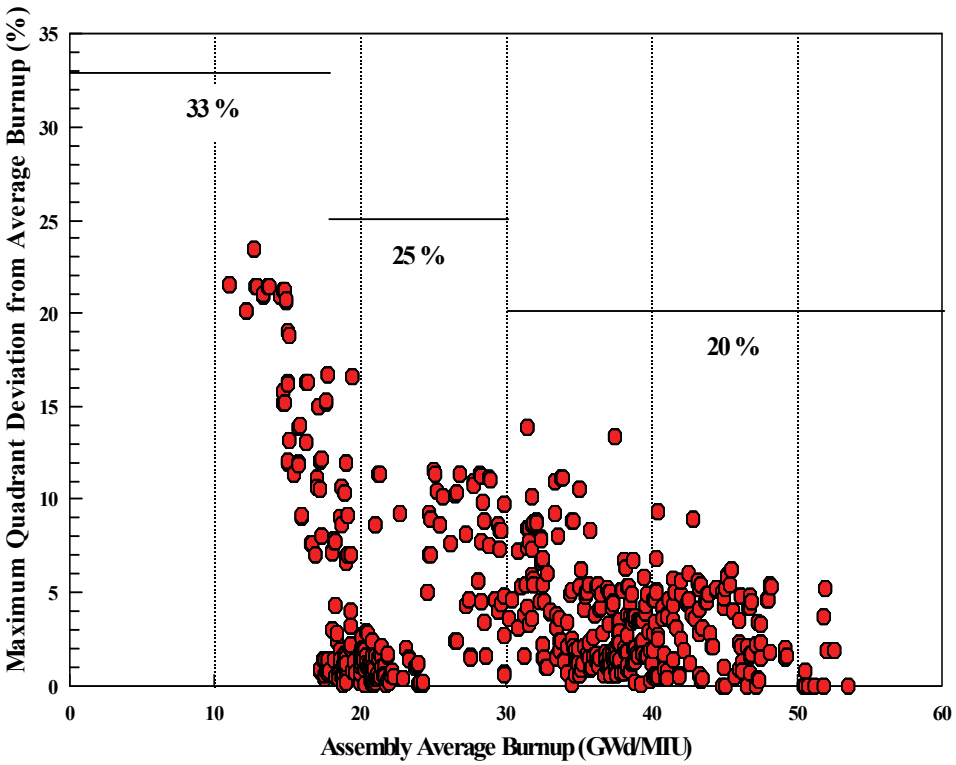

Figure 13. Selection of Limiting Horizontal Burnup Profiles [18]

IMPLEMENTING BURNUP CREDIT

With the computer codes validated and limiting conditions for depletion and modeling of the cask/rack selected, the loading curves are generated. Figure 1 is an example of a typical loading curve; however, more complexity than that represented by Fig. 1 is generally utilized in

developing a set of loading curves. In spent fuel pool analyses separate loading curves are given for different possible arrangements of the fuel, e.g., uniform loading of the rack, checkerboard loading, one empty cell out of every four, a control rod in one of every four. For casks, separate loading curves are given for the different fuel types that can be loaded in the cask and in some cases different loading arrangements. Since use of multiple loading curves is possible, careful analysis is required for any interfaces between loading patterns. An arrangement of fuel of different enrichments meeting the loading curve does not represent an interface, but interfaces between assemblies with different water fractions are a concern; a safety margin for the interface must be included.

The loading curves specify a minimum burnup for a given enrichment. The reactor record assembly burnup and enrichment is compared to these curves to determine if loading the assembly is permitted. The reactor record burnup has an uncertainty from both power sharing and absolute core power level uncertainties. A value of about 5% of the burnup of record has generally been used. In pool burnup credit this uncertainty can be imbedded in the loading curves with other uncertainties pertaining to fuel and rack manufacturing tolerances, isotopic compositions, and fuel positioning. The uncertainty in the burnup of record can also be subtracted from the burnup of record before it is compared to the limit. A third option is to increase the burnup limit by the appropriate uncertainty. Since the reactor record uncertainty can be plant dependent, the loading curves for casks do not include the reactor record uncertainty. In that case the plant must decrease its reactor record burnup or increase its required burnup by the uncertainty prior to comparing to the loading curve.

To verify there are no errors in the reactor records for the burnup of the assembly, a burnup verification measurement may be required. The International Atomic Energy Agency (IAEA) currently requires a burnup verification measurement for shipping casks [19]. The NRC recommends a verification measurement for burnup credit shipping casks [20]. Even though the NRC recommends a verification measurement of the reactor burnup record before a burnup credit cask is shipped, utilities are loading casks as storage canisters without a burnup verification measurement. No verification measurement is needed for spent fuel pool applications in the United States.

There are three burnup measurement devices that have been used to measure burnup of actual spent fuel: the Fork detector (and the Fork Plus), the BNFL detector, and the PYTHON. When appropriately calibrated, these methods produce similar measurement accuracies for assemblies that are similar to the calibration assembly, which is about 5% of the burnup value. ORNL presents a review of burnup confirmation measurements in Reference 21.

BURNUP CREDIT APPLICATIONS

Burnup credit has been used for spent fuel pools, spent fuel storage and transportation casks, and reprocessing. To assure consistent and safe burnup credit analysis for these applications a standard (ANSI/ANS-8.27) has been written [22].

Spent Fuel Pool Applications

The first application of burnup credit was for PWR spent fuel pools. At this point all PWR spent fuel pools in the United States use burnup credit. In the rest of the world there are often non-burnup credit spent fuel pools. The key reason for this is that reprocessing in these countries reduces the need to increase pool storage.

All BWR pools in the United States utilize Gd credit, which is a form of burnup credit. Gd credit is also used in spent fuel pools in Europe. BWR pools have not used burnup credit for the reduction in reactivity beyond the peak reactivity.

Cask Applications

Burnup credit is employed to determine loadings for PWR storage casks. This burnup credit is used for borated pools and is very similar to spent fuel pool burnup credit. Alternatively, significant credit for the soluble boron is used to establish loadings for the cask.

In the 1990s DOE supported an effort to license burnup credit for casks. As part of this effort a topical report was prepared [23]. Although never approved, the topical report is a good reference for burnup credit requirements. The NRC did, however, use the topical report experience to issue guidance for burnup credit applications for transportation casks [20]. No current cask licenses actually follow this guidance. In 2006 the NRC issued its first burnup credit license. As of 2010 there is still only one transport cask for use in the United States that has received a burnup credit license. The burnup credit license includes use of credit for actinides and fission products. Details of the methods are proprietary. Many PWR spent fuel storage casks are also currently using burnup credit. The NRC has also licensed a transport cask that uses actinide-only burnup credit, and Swiss regulators have approved it for use in Switzerland. Burnup credit has been used to raise the enrichment limit of a transport cask used in Europe. Only limited burnup credit was needed for this cask so a very conservative approach was taken.

The preferred storage systems for PWR spent fuel are dual purpose (storage and transport) canister systems. Most of these new systems depend on burnup credit for transport even though only one system has received its transport license. The burnup credit analysis is accepted for storage, though questions remain about transporting these dual purpose canisters as currently loaded.

Other Applications

Burnup credit has also been used for disposal and reprocessing. In disposal, burnup credit is used to analyze flooded casks after repository closing. Since the consequences of a criticality accident may be small, a probabilistic approach can be used that is less conservative in some respects than some of the treatments discussed in this module. In reprocessing, burnup credit has been used to reduce the amount of absorbers required to be added to the dissolver tanks.

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ACRONYMS

BA	Burnable Absorber.
BAA	Burnable Absorber Assemblies
BNFL	British Nuclear Fuels Limited
BP	Burnable Poison – This is the same as BA.
BWR	Boiling Water Reactor
DOE	U. S. Department of Energy
EPRI	Electric Power Research Institute
GWD/MTU	GigaWatt-Days per Metric Tonne of initial Uranium.
IAEA	International Atomic Energy Agency
IFBA	Integral Fuel Burnable Absorber. This is any burnable absorber where the absorbing material is inside the fuel clad.
MOX	Mixed Oxide fuel. This fuel is plutonium and uranium oxide.
MW/MTU	MegaWatt per Metric Tonne of initial Uranium. This is a unit of specific power.
NEA	Nuclear Energy Agency of OECD.
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development.
OFA	Optimized Fuel Assembly. A fuel design by Westinghouse with a smaller pin diameter than that of standard assemblies.
ORNL	Oak Ridge National Laboratory.
pcm	Percent millirho (10^{-5} in delta-k).
ppm	Parts per million by weight.
PWR	Pressurized Water Reactor
SCALE	Standardized Computer Analyses for Licensing Evaluation. This is the name of a computer code system used for criticality safety analysis.
SFCOMPO	Spent Fuel Isotopic Composition Database
USL	Upper Subcritical Limit on k_{eff} to assume safety.
WABA	Wet Annular Burnable Absorber

PROBLEMS

1) The calculated k of a cask is 1% higher than the upper subcritical limit. Approximately how much additional assembly average burnup is required?

a) Assuming full burnup credit.
b) Assuming actinide-only burnup credit.

2) A depletion analysis was done assuming a constant 800 ppm of soluble boron. It was noticed that for an unfortunate set of assemblies the cycles were shut down early and the actual average concentration they saw was 900 ppm. About how much more burnup should be required for these assemblies?

a) Assuming full burnup credit.
b) Assuming actinide-only burnup credit.

3) A cask is designed for BWR fuel that uses Gd credit at burnup values out to peak reactivity. What is the appropriate burnup verification measurement for loading this cask?

4) For a low-burned assembly it is usually conservative to use an axially flat burnup distribution. Why?

5) If you decided to have a depletion analysis for every burnup needed, how many depletion analyses would be needed to create a final loading curve using points at 5 different enrichments and using the 18-axial and 2-horizontal zone DOE burnup distributions?

SOLUTIONS

1a) With full burnup credit the change in k with burnup is about 1% per GWD/MTU. The added burnup requirement however comes with additional uncertainties. There is a 5% of the delta-k of depletion uncertainty and a 5% burnup record uncertainty. Therefore the added burnup needed is 1.1 GWD/MTU plus or minus about 0.3 GWD/MTU.

1b) With actinide-only burnup credit the change in k with burnup is about 0.5% per GWD/MTU (See Figure 4). The added burnup requirement would be 2 GWD/MTU plus the uncertainty change. In actinide-only burnup credit the depletion bias and uncertainty is generally a slight function of burnup. Thus the additional bias and uncertainty is negligible. The burnup record uncertainty is still 5% of the burnup. The total added burnup is about 2.1 GWD/MTU. This approximation is plus or minus about 0.5 GWD/MTU.

2) The depletion analysis was done with 100 ppm less soluble boron. This module gives a sensitivity of 3 to 3.5 pcm/ppm. Thus the low ppm in the depletion is non-conservative by about 350 pcm which is 0.35% in k.

a) For full burnup credit this means about 0.35 GWD/MTU plus 10% (see solution to problem 1a) needs to be added. Thus 0.38 GWD/MTU additional burnup should be required.
b) For actinide-only burnup credit about 0.7 GWD/MTU plus 5% (see solution to problem 1b) needs to be added. Thus 0.73 GWD/MTU needs to be added.

3) In Gd credit the most reactive condition for all burnups is used. Therefore, Gd credit does not depend on the assembly burnup. Since there is no dependence on the assembly burnup, verification of the assembly burnup is of no value and no measurement should be performed.

4) The fuel burnup is greater in the center of the fuel. At low burnups the flux in the cask or rack is center-peaked where there is the most burnup. A uniform burnup assumption reduces the burnup at the center and adds it to the ends. So long as the flux is center-peaked the uniform burnup assumption is conservative. With higher burnup, the reactivity of the ends is so much greater than the center (since the ends have less burnup) the flux shape starts to be top-peaked. With the top-peaked flux the uniform burnup assumption, which effectively moves burnup from the center to the ends, now is non-conservative.

5) (18 axial nodes) x (2 horizontal nodes) x (5 enrichments) = 180 depletion analyses