BURNUP CREDIT APPROACH USED IN THE YUCCA MOUNTAIN LICENSE APPLICATION

J.M. Scaglione, J.C. Wagner

Oak Ridge National Laboratory[*]
Nuclear Science and Technology Division
P.O. Box 2008, Bldg. 5700
Oak Ridge, TN 37831-6170 USA
E-mail: scaglionejm@ornl.gov

Abstract—The United States Department of Energy has submitted a license application (LA) for construction authorization of a deep geologic repository at Yucca Mountain, Nevada. The license application is currently under review by the United States Nuclear Regulatory Commission (NRC). This paper will describe the methodology and approach used in the LA to address the issue of criticality and the role of burnup credit during the postclosure period. The most significant and effective measures for prevention of criticality in the repository include multiple redundant barriers that act to isolate fissionable material from water (which can act as a moderator, corrosive agent, and transporter of fissile material); inherent geometry of waste package internals and waste forms; presence of fixed neutron absorbers in waste package internals; and fuel burnup for commercial spent nuclear fuel. A probabilistic approach has been used to screen criticality from the total system performance assessment. Within the probabilistic approach, criticality is considered an event, and the total probability of a criticality event occurring within 10,000 years of disposal is calculated and compared against the regulatory criterion. The total probability of criticality includes contributions associated with both internal (within waste packages) and external (external to waste packages) criticality for each of the initiating events that could lead to waste package breach. The occurrence of and conditions necessary for criticality in the repository have been thoroughly evaluated using a comprehensive range of parameter distributions. A simplified design-basis modeling approach has been used to evaluate the probability of criticality by using numerous significant and conservative assumptions. Burnup credit is used only for evaluations of in-package configurations and uses a combination of conservative and bounding modeling approximations to ensure conservatism. This paper will review the NRC regulatory criteria relevant to postclosure criticality, explain the role of criticality within the overall repository performance assessment, describe the strategy for preventing criticality via design features and waste form properties, and discuss the numerous considerations relevant to criticality and burnup credit for spent nuclear fuel disposed of in a geologic repository, with emphasis on the burnup credit approach and analyses.

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1. Background

The U.S. Department of Energy (DOE) submitted a license application [1] in June of 2008 to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a high-level waste geologic repository in accordance with Title 10 of the Code of Federal Regulations (CFR) Part 63. The license application was subsequently accepted for review by the NRC in September of 2008 under Docket Number 63-001. The postclosure nuclear criticality analysis methodology followed the risk-informed, performance-based process presented in *Disposal Criticality Analysis Methodology Topical Report* [2]. This methodology included taking credit for the reduced reactivity potential of irradiated commercial light-water reactor fuel assemblies in criticality analyses (i.e., burnup credit). The DOE used a total system performance assessment (TSPA) to demonstrate compliance with the postclosure individual protection standards of proposed 10 CFR 63.311[*] and 10 CFR 63.321 and the groundwater protection standards of 10 CFR 63.331. Within this assessment, features, events, processes, and sequences of events and processes that might affect the Yucca Mountain repository were examined. Criticality was screened from consideration in the TSPA in accordance with the probability criterion in proposed 10 CFR 63.342(a),[*] which states: "DOE's performance assessments conducted to show compliance with 63.311(a)(1), 63.321(b)(1), and 63.331 shall not include consideration of very unlikely features, events, or processes, i.e., those that are estimated to have less than one chance in 10,000 of occurring within 10,000 years of disposal."

Characterizing the nuclear data and the depletion/criticality modeling capabilities of different codes is an important part of using burnup credit, but the sensitivity of these characterizations is largely dependent upon the specific application (i.e., spent fuel packaging strategy). The spent nuclear fuel (SNF) disposal strategy includes using transportation, aging, and disposal (TAD) canisters to transfer the majority of the waste from the generating sites to the geologic repository operations area. TAD canisters would be loaded with commercial SNF at the respective power facilities according to loading curves. The TAD canisters would then be placed inside a waste package prior to disposal. The waste package design consists of two concentric cylinders—an inner vessel made of Stainless Steel Type 316 (UNS S31600), modified with additional constraints on the nitrogen and carbon content, and an outer corrosion barrier made of Alloy 22 (UNS N06022) (with some restrictions on the range of alloying elements), a corrosion-resistant nickel-based alloy. Waste packages in emplacement drifts would have titanium drip shields to protect them from dripping water and rockfall during the postclosure period.

The criticality calculations included the use of burnup credit for 29 principal isotopes (14 actinides and 15 fission products) as presented in Table 1. Credit for burnup was used only for in-package considerations; external (out-of-waste package) criticality evaluations were based on a fresh fuel assumption and are therefore not discussed further in this paper.

Computational biases and uncertainty were established using a combination of publicly available information and proprietary data for benchmarks, including commercial reactor criticals, laboratory critical experiments, and radiochemical assay data. Destructive analysis results were used to determine isotopic concentration bias, and critical configurations were used to determine bias in k_{eff} predictions. Criticality evaluations were performed using standard, well-established computer codes—MCNP [3] for criticality analyses and SCALE/SAS2H [4] for depletion calculations.

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[*]On March 13, 2009, the NRC final rule implementing revised standards of the U.S. Environmental Protection Agency was published in the Federal Register. See Implementation of a Dose Standard After 10,000 Years, 74 Fed. Reg. 10,811. The final rule became effective April 13, 2009, and does not contain any material differences from the proposed rule with respect to this paper.

Table 1. Principal Isotopes for Disposal Burnup Credit

Fission Products				
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh	^{109}Ag
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm
^{151}Sm	^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd
Actinides				
^{233}U	^{234}U	^{235}U	^{236}U	^{238}U
^{237}Np	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu
^{242}Pu	^{241}Am	^{242m}Am	^{243}Am	—

The burnup credit methodology employed two separate calculational models—an isotopic model for performing the fuel irradiation analyses and a criticality model for performing the criticality evaluations. These models were used for developing penalty factors in terms of Δk_{eff} consistent with ANSI/ANS-8.17 [5], for establishing an allowable neutron multiplication factor. Consistent with ANSI/ANS-8.1 [6] and ANSI/ANS-8.17 [5], biases and uncertainties have been developed for both the isotopic depletion calculations and the criticality calculations for use in establishing the critical limit (CL). The CL is characterized by statistical tolerance limits that account for biases and uncertainties associated with the criticality code (i.e., the determination of the lower bound tolerance limit [LBTL]) and any uncertainties due to extrapolation outside the range of experimental data, as well as limitations in the geometrical or material representations used in the computational method. The CL was calculated using Equation 1.

$$CL(x) = f(x) - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m \quad \text{(Eq. 1)}$$

where

x = a neutronic parameter used for trending

$f(x)$ = the LBTL function accounting for biases and uncertainties that cause the calculation results to deviate from the true value of k_{eff} for a critical experiment, as reflected over an appropriate set of critical experiments

Δk_{EROA} = penalty for extending the range of applicability

Δk_{ISO} = penalty for isotopic composition bias and bias uncertainty

Δk_m = traditional administrative margin to ensure subcriticality

The range of applicability of the benchmark experiments covered the range of parameters of the application model; hence, the Δk_{EROA} term was zero. In contrast to "traditional" nuclear criticality safety analyses and associated governing regulations, in which the purpose is to ensure prevention of criticality and corresponding protection of personnel and facilities, the purpose of the postclosure criticality evaluation is to determine the probability of a criticality event in the postclosure time period to establish whether it should be included in the performance assessment. Hence, the Δk_m term was assigned a value of zero for the licensing basis. To ensure that the calculated k_{eff} value is always greater than the actual k_{eff} value, conservative assumptions and modeling representations were used to define the design-basis configurations in performing the criticality calculations for each configuration. Collectively, these assumptions and representations result in overestimation of the k_{eff} value and provide margin in the analysis predictions and loading curves. Therefore, there is margin in the evaluations of k_{eff} and that margin is sufficient to ensure that there is a high degree of confidence that configurations determined to be subcritical are so.

The criticality calculations were ultimately used in a probabilistic assessment for calculating the probability of one or more criticalities occurring in the repository over the first 10,000 years after closure. To ensure that a criticality event does not occur in the repository, the TAD canister performance specifications [7] were selected such that the initial emplaced configuration for all commercial SNF remains subcritical, even under flooded and degraded conditions. Therefore, a deviation (human factor error) in either the as-designed properties/specifications or the waste loading that would result in an increase in reactivity must occur in order to achieve criticality. A design-basis modeling approach was used to simplify and bound the probability of criticality calculation.

2. Neutronic Analyses

2.1 Isotopic Analyses

The isotopic analyses used the SAS2H control module of the SCALE code system to simulate isotopic changes as fuel is irradiated in a reactor. Isotopic concentrations were calculated for the principal isotopes listed in Table 1. These isotopic concentrations were then utilized in the criticality models for commercial SNF. A bias, in terms of Δk, was determined for the set of principal isotopes, based on comparisons between calculated and measured data. The difference (Δk) is a direct measure of the net bias and uncertainty in the k_{eff} calculation due to the variability in the predicted nuclide concentrations. This method, referred to as the "direct-difference" method, evaluates the aggregate effect of the nuclide uncertainties on k_{eff} directly and does not require a statistical analysis of bias and uncertainty for any individual nuclide. Rather, the net effect of bias and uncertainty from all nuclides is determined directly from analysis of the mean and variance of the distribution of Δk_{eff} values obtained using the predicted and measured nuclide concentrations from many experiments. Reference [8], Sections 5.1.5 and 6.1, demonstrates that the direct-difference method produces bias and bias uncertainty values similar to those of other best-estimate methods, such as Monte Carlo uncertainty sampling and sensitivity and uncertainty methods. The best-estimate methods allow random uncertainties in the calculated nuclide concentrations to produce partially compensating reactivity effects (e.g., the reactivity effect of overpredicted concentration for a fission product may be partially "canceled out" by the reactivity effect of underpredicted concentration for another fission product). Best-estimate methods are considered more realistic methods of estimation compared with the "bounding method," which is highly conservative due to its use of isotopic composition uncertainties in such a way as to maximize their effects on system reactivity in a nonphysical way. The isotopic prediction bias and uncertainty are represented in Equation 1 as the term Δk_{ISO}.

The validation of the isotopic (SAS2H) calculations considered commercial reactor critical (CRC) and radiochemical assay (RCA) data from both pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) to determine the bias and bias uncertainty in calculated k_{eff} values associated with the computed isotopic compositions. CRC data represent an integral effect over a large complex system and were not used explicitly for calculating isotopic bias, but rather to demonstrate the combined integral effects of isotopic and criticality calculation bias and bias uncertainty on calculated k_{eff} values. The bias and bias uncertainty of the CRC and RCA data k_{eff} values were evaluated in *Isotopic Model for Commercial SNF Burnup Credit* [9]. The overall integral reactivity bias for the CRC data was quantified by calculating k_{eff} between the measured (always 1.0) and calculated k_{eff} for each of 57 (41 PWR, 16 BWR) CRC cases. For the RCA data, the bias and uncertainty in k_{eff} values was established by comparing criticality calculations performed using measured isotopic concentrations from 104 (74 PWR, 30 BWR) assay samples with calculations performed using SAS2H depletion code calculated isotopic concentrations for the assay samples. The bias and bias uncertainty values based on the commercial reactor critical data and the measured radiochemical assay data were predicted to be −0.0077 and −0.0249 Δk, respectively [9]. The RCA data isotopic bias penalty factor was used for the Δk_{ISO} term in Equation 1 because it is more limiting. A confidence level of 95% was used in calculating the lower bound for the tolerance limit, which covers 95% of the population for each data set. Note that the large bias and bias uncertainty for the radiochemical assay data is primarily a result of the uncertainty associated with the RCA data, which manifests itself as a higher penalty in the tolerance limit.

2.2 Criticality Analyses

The criticality analyses used the general-purpose Monte Carlo N-Particle (MCNP) transport code to analyze the geometry and materials that define a configuration and to calculate the effective neutron multiplication factor (k_{eff}). The nuclear cross-section data distributed with MCNP 5.1.40 and used to model the various physical processes are based primarily on the Evaluated Nuclear Data File/B Version VI (ENDF/B-VI) library. The criticality model was validated so that the range of applicability covers the various configurations of intact and degraded fuel that could occur in the repository over long time periods. The criticality model validation was documented in *Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel* [10].

A key element in all criticality calculations is the geometric configuration of the fissile material. Since long-term geologic storage must consider other parameters affecting the repository that are temporally and spatially dependent, a number of waste package internal configurations are possible over time. As the repository, waste package, and TAD designs evolved, comprehensive criticality evaluations were performed to determine the configurations and parameters of influence that yield the highest system k_{eff} values over the first 10,000 years after waste emplacement. These evaluations led to the development of models (i.e., design-basis configurations) to bound potential relevant variations in materials, geometry, and neutron spectrum that occur as the internals of the SNF canisters change over long time periods. The most reactive configurations (design-basis configurations) were developed considering processes that result in maximizing k_{eff} while accounting for repository characteristics, material characteristics of the waste forms and basket structures, and chemical and physical mechanisms for internal reconfiguration.

The following provides a listing of conservative modeling parameters used in the criticality model forming the design basis:

(1) Most-reactive fuel assembly design used (GE 7 × 7 for BWRs and B&W 15 × 15 for PWRs [11, Attachment II])—Use of most-reactive fuel assembly design simplifies the licensing basis by providing a bounding value for comparison with other assembly designs.
(2) Commercial SNF stack density at 98% of theoretical density after irradiation—Selected because it increases equivalent fuel loading and is expected to bound future higher-density fuel.
(3) Neutron absorber plate thicknesses accounting for 10,000 years of general corrosion—This reduces the amount of neutron absorber material interstitial to the assemblies and results in a reduction in assembly pitch (i.e., increases assembly-to-assembly interaction).
(4) Neutron absorber plates credited with 75% of design-specified absorber material—Selected consistent with existing NRC guidance [12, 13] on use of fixed absorbers and to bound uses where 90% credit is obtained.
(5) Fuel isotopic composition represented at 5-year cooling time—Selected because it provides the highest reactivity over the relevant range (5 to 10,000 years). The conservatism of the 5-year cooling time compositions to be used to represent SNF compositions for the postclosure period has been demonstrated in Ref. [14], Section 2.2.
(6) Fuel depletion parameters selected to increase residual reactivity at discharge (see Section 3.2).
(7) Conservative representation of axial burnup profile for PWR loading curves (BWR burnup credit required is too low for axial effects to become pronounced)—Parallel calculations are performed using a single-zone uniform axial burnup distribution and a multizone limiting axial burnup profile [20]. The more limiting of the two is used for setting the criticality safety requirements. Burnup credit that includes fission products typically produces higher k_{eff} values

at lower burnups (<~40 GWd/MTU) when using a uniform burnup profile than when using conservative axial burnup profiles.

(8) Internal geometry represented as tight-packed cylindrical array with reduced-width absorber plates increasing neutronic interaction rates—This model bounds other degraded configurations. The tight-pack geometry is bounding as it disregards the physics associated with oxidation of interstitial materials. As materials oxidize they expand and act to displace the moderator. In addition, this phenomenon also results in a larger assembly-to-assembly pitch. Both of these phenomena act to decrease system reactivity.

(9) Calculations are based on fully flooded systems—One of the most important parameters for criticality evaluations is the amount of water that can enter and stay inside a waste package to moderate neutrons. Because this value can vary depending on waste package breach initiating events (e.g., seismic vibratory-induced failure), the design-basis configuration assumes that any crack in the waste package outer barrier results in a fully flooded package. No credit was taken for the engineered barrier system drip shield diverting seepage water, evaporative processes (if liquid water did enter the package), or the waste package inner vessel and TAD canister acting as barriers to water ingress.

(10) Maximum reflector effectiveness (dry tuff surrounding package)—This is bounding because a waste package that is filled with water would most likely have wet material outside the package. The dry tuff results in lower neutron thermalization of neutrons that exit than when using wet tuff, thereby resulting in a slight increase in the number of neutrons that reenter the waste package.

(11) No credit taken for moderator displacement effect of corroded neutron absorber plates or other internal components—This is a simplifying assumption and is bounding because the waste package configuration is an undermoderated system and pure water is a better moderator than oxidized and hydrated corrosion products.

Radial burnup profiles were considered for inclusion in the design-basis configuration. A PWR radial burnup gradient reactivity evaluation was performed in Reference [15], where it was concluded that the change in k_{eff} due to radial burnup gradients is expected to be inconsequential to system reactivity.

Numerical codes must be validated against benchmark experiments with characteristics similar to the identified application. The criticality model validation process was performed for the range of parameters and conditions defining the design-basis configuration. Applicable critical experiments based on neutronic similarity between the design-basis system and selected critical experiments were determined using sensitivity/uncertainty (S/U) analysis methods. S/U analysis methods can be used to demonstrate that nuclear systems with similar physical characteristics, including material compositions, geometry, and neutron flux spectra, exhibit similar sensitivities of k_{eff} to perturbations in the neutron cross-section data on an energy-dependent, nuclide-reaction-specific level. The critical experiments that were evaluated for applicability included publicly available mixed-oxide (PuO_2 and UO_2) and low-enriched uranium critical experiments from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* [16], proprietary HTC mixed-oxide critical experiments [17], and CRCs. HTC refers to "Haut Taux de Combustion," which is a French designation for "high burnup." The 156 HTC critical experiments were performed in France with fuel pins having uranium and plutonium isotopic compositions that were designed to be similar to PWR fuel that had an initial enrichment of 4.5 wt % ^{235}U burned to 37,500 MWd/MTU.

MCNP does not have an S/U capability. Therefore, the S/U analysis was performed with the TSUNAMI-3D sequence and the TSUNAMI-IP module distributed as part of the SCALE 5.1 code package and was used to quantify the applicability of the experiments to the design-basis configuration. TSUNAMI-3D is a Monte Carlo–based eigenvalue sensitivity analysis sequence. This software tool calculates energy-, mixture-, nuclide-, and region-dependent sensitivity of the system k_{eff} to variations in nuclear data of modeled materials. TSUNAMI-3D uses first-order linear-perturbation theory to calculate sensitivity coefficients. CENTRMST/PMCST/WORKER and the SCALE ENDF/B-VI 238-group cross-section library were used in cross-section resonance processing for all cases except for the CRC calculations, which used NITAWLST and the SCALE ENDF/B-V 238-group cross-section library. To increase confidence that the sensitivity profiles were accurate, direct perturbation calculations were performed for selected cases to verify that the TSUNAMI-3D–calculated sensitivity coefficient values were consistent with the sensitivity coefficients predicted via direct perturbation calculations with MCNP. Overall, the two methods were considered to be in good agreement [10].

A set of indices has been defined for use in S/U analyses that provides a measure of the neutronic similarity between a design system and a critical experiment. The TSUNAMI-IP code utilizes sensitivity data developed for benchmark experiments and for an identified application along with the cross-section covariance data to numerically quantify the similarity of a benchmark to the identified application. A widely used index for similarity assessment is the correlation of k_{eff} uncertainties, known as c_k. The c_k index quantifies the amount of shared uncertainty in the k_{eff} values of an application and a benchmark due to cross-section uncertainties. Integral index c_k is the correlation coefficient between sensitivity-weighted uncertainties in the application system and in an experiment system. A c_k value of 0.0 represents no correlation between the systems, a value of 1.0 represents full correlation between the systems (i.e., identical systems), and a value of –1.0 represents a full anticorrelation.

Published guidance for similarity criteria based on experience at Oak Ridge National Laboratory [18, 19] was used to identify critical experiments applicable to the design-basis configuration. A critical configuration was considered applicable to an evaluation case if the c_k value was $\geq$ 0.9, a critical configuration was considered marginally applicable if c_k was $\geq$ 0.8 and < 0.9, and a critical configuration was considered not applicable if c_k < 0.8. Reference [18] recommends that the validation methodology should include about 15 to 20 very correlated systems (c_k of 0.90 or higher) or 25 to 40 moderately correlated systems (c_k of 0.80 or higher). A plot showing the c_k values and illustrating the applicable critical experiments to bias and bias uncertainty calculations for the 21-PWR waste package for commercial SNF of 5.0 wt % ^{235}U initial enrichment at 40 GWd/MTU burnup is provided in Fig. 1. A summary listing of the critical experiments found to be applicable to the systems of interest is found in Table 2.

The LBTL function (criticality code bias and bias uncertainty) was established using only applicable benchmark critical experiments (i.e., critical experiments having neutronic similarities with the application system [$c_k \geq 0.8$]), which also prescribe the basic range of applicability of the results. For the commercial reactor criticals, an uncertainty of 2% (2 standard deviations at 95% confidence level) in k_{eff} was used to account for uncertainties in the commercial reactor critical configurations [10, Section 6.3.3]. The LBTLs are presented in Table 3 [10].

Table 2. Summary of Applicable Critical Experiments to Bias and Bias Uncertainty Determination

Application system		Number of applicable critical experiments					
Waste package	**Enrichment (wt % ^{235}U)/ burnup (GWd/MTU)**	**MOX lattice**	**MOX solution**	**LEU**	**HTC**	**CRC**	**Total**
21-PWR nominal configuration	2.0/ 0	0	0	37	0	4	41[a]
	3.0/ 0	0	0	37	1	5	43[a]
	3.0/ 15	17	0	0	145	56	218
	3.5/ 25	18	0	0	145	56	219
	4.0/ 30	18	1	0	145	56	220
	4.5/ 35	19	11	0	145	56	231
	5.0/ 40	18	1	0	145	56	220
21-PWR design-basis configuration	2.0/ 0	0	0	37	0	4	41[a]
	3.0/ 0	0	0	37	0	5	42[a]
	3.0/ 15	17	0	0	145	56	218
	3.5/ 25	18	0	0	145	56	219
	4.0/ 30	18	1	0	145	56	220
	4.5/ 35	19	2	0	145	56	222
	5.0/ 40	18	1	0	145	56	220
44-BWR nominal configuration	3.0/ 0	0	0	37	0	4	41[a]
	3.0/ 10	20	9	0	145	56	230
	4.0/ 0	0	0	37	0	4	41[a]
	4.0/ 20	21	17	0	145	51	234
	5.0/ 30	21	19	0	145	51	236
44-BWR design-basis configuration	3.0/ 0	0	0	37	0	4	41[a]
	3.0/ 10	19	9	0	145	56	229
	4.0/ 0	0	0	37	0	4	41[a]
	4.0/ 20	21	17	0	145	51	234
	5.0/ 30	21	21	0	145	51	238

[a]Only the applicable LEU LCEs with EALF values below 0.3882 eV (36 LEU LCEs) were used to calculate the LBTL.

Table 3. LBTL and Corresponding Range of Applicability

Application systems	LBTL[a]	Range of applicability; *EALF* in eV[b]
21-PWR waste packages containing fresh fuel	0.9905	$0.0977 \leq EALF \leq 0.3882$
21-PWR waste packages containing burned fuel	0.9778	$0.0684 \leq EALF \leq 1.0410$
44-BWR waste packages containing fresh fuel	0.9905	$0.0977 \leq EALF \leq 0.3882$
44-BWR waste packages containing burned fuel	0.9778	$0.0421 \leq EALF \leq 0.9679$

[a]Fraction of the k_{eff} population above the LBTL value is 95%; the confidence on population is 95%.

[b]This column shows the *EALF* range for the critical experiments used to determine the single-valued LBTL function.

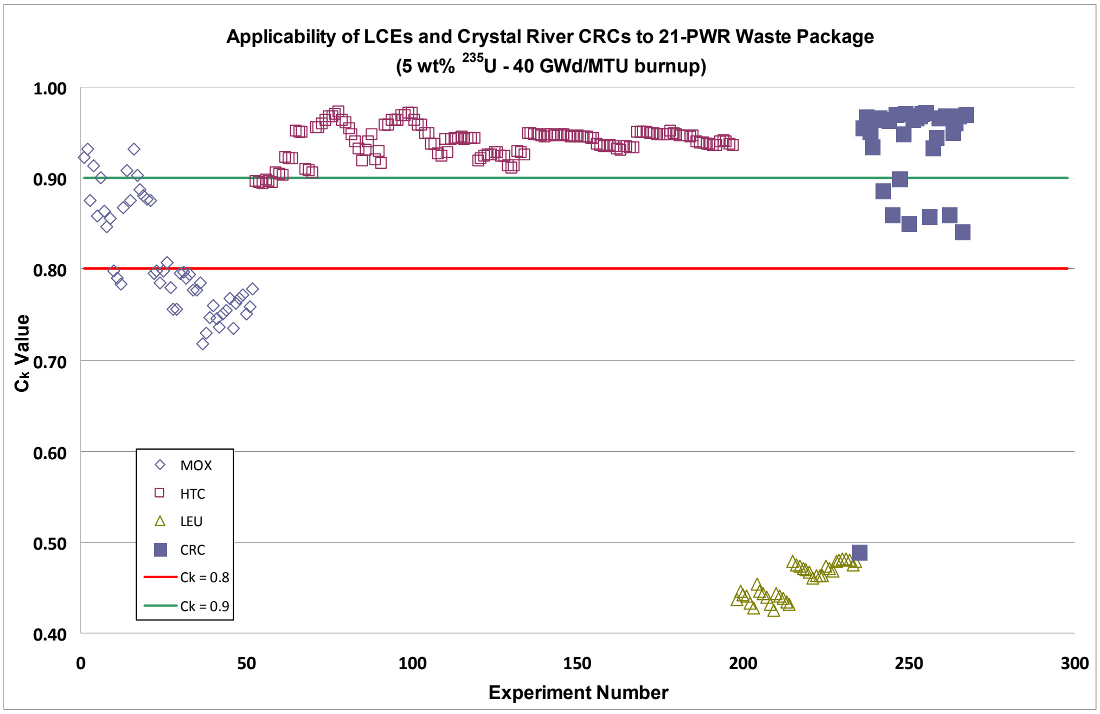

Figure 1. Value for c_k as a Function of Experiment Number for 21-PWR Waste Package with Commercial SNF at 5 wt % ^{235}U Initial Enrichment and 40-GWd/MTU Burnup.

3. Application of Burnup Credit in the Licensing Basis

The application of burnup credit for spent fuel disposal is coalesced in the generation of loading curves. Loading curves, which are functions of burnup and enrichment, are the loci of values delineating the region of acceptable burnup/enrichment combinations for criticality control. Criticality loading curves were established such that the k_{eff} of a waste package fully loaded with assemblies selected from the curve will be less than the critical limit under postulated postclosure conditions. The criticality loading curves were used in probabilistic evaluations to calculate the probability of criticality as a result of a misload (i.e., not loading according to the loading curves). The loading curves were generated using the design-basis configurations and obtained by determining the burnup value at which the waste package $k_{eff} + 2\sigma$ value is equal to the critical limit. By using the design-basis configuration, the loading curve is generated once and the assigned burnup values of all assemblies considered for loading into a TAD canister are compared directly against this loading curve. It is recognized that this can result in significant conservatism for reactor sites that have assembly designs different from the design basis. However, the licensing basis was developed to accommodate multiple canister criticality control design configurations because a licensed TAD canister design does not currently exist. Using the design basis for all TADs enables a bounding probability of criticality occurrence in the repository to be calculated. The process for developing the criticality loading curves for each canister configuration and range of commercial SNF characteristics is documented in *CSNF Loading Curve Sensitivity Analysis* [20]. The critical limit used for generating the loading curves, calculated using Equation 1 with the terms discussed above, for the PWR and BWR commercial SNF was 0.9529 [1, Table 2.2-11].

The loading curves for the PWR and BWR TAD canisters are presented in Fig. 2 with a discretized representation of the spent fuel inventory superimposed on the respective loading curves. Here, the number of assemblies falling within 0.1 wt % enrichment and 1-GWd/MTU burnup bins are shown in a

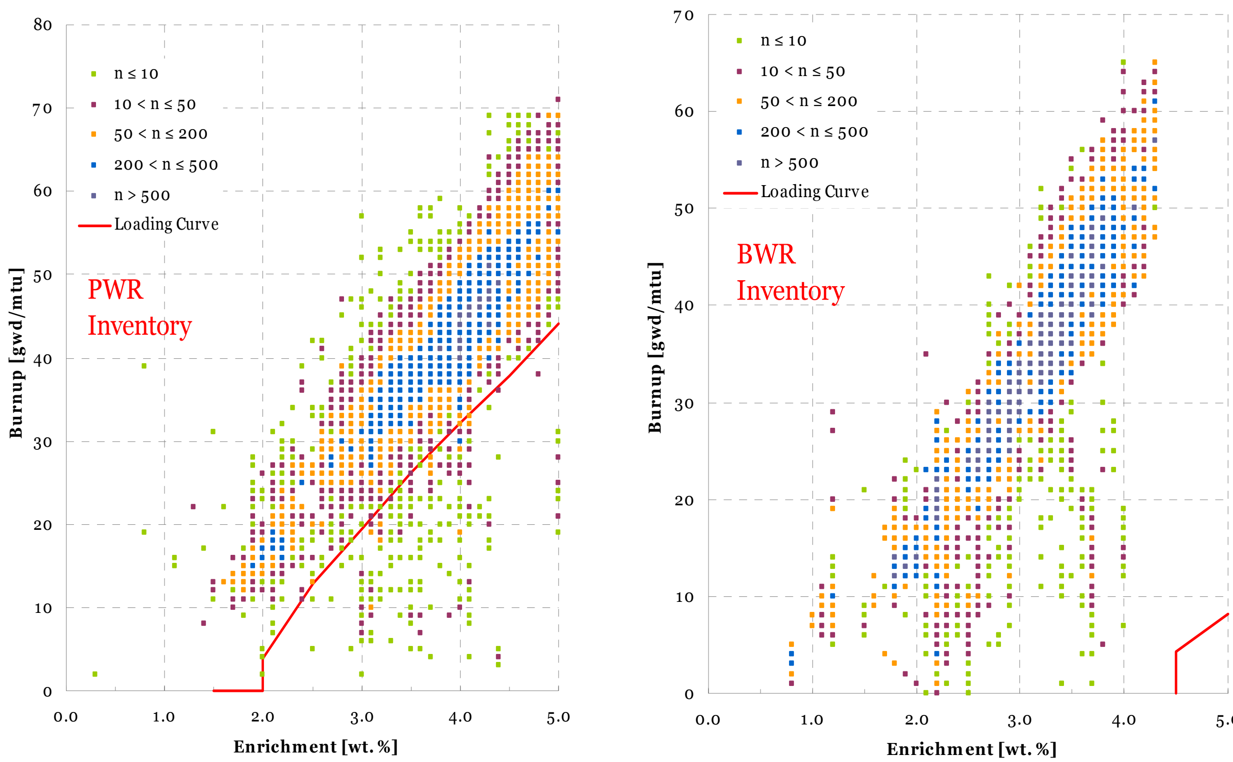

Figure 2. CSNF Loading Curves.

color-coded format. The color indicates the number of assemblies with burnup and enrichment falling within each bin, as indicated in the legend.

The PWR loading curve is characterized by the following equation:

$$b = \begin{cases} 0 \text{ for } e < 2.0 \text{ wt } \% \\ 0.4854e^5 - 8.6621e^4 + 60.9498e^3 - 211.9900e^2 + 378.3106e - 269.4040 \\ \text{for } 2.0 \text{ wt } \% \le e \le 5.0 \text{ wt } \% \end{cases} \qquad \text{(Eq. 2)}$$

The BWR loading curve is characterized by the following equation:

$$b = \begin{cases} 0 \text{ for } e < 4.5 \text{ wt } \% \\ 7.7911e - 30.7201 \text{ for } 4.5 \text{ wt } \% \le e \le 5.0 \text{ wt } \% \end{cases} \qquad \text{(Eq. 3)}$$

The criticality model for generating the loading curves was based on all assemblies characterized by the conservative modeling representations (i.e., modeling representations that increase the calculated k_{eff} value) described above and having burnup and enrichment combinations that correspond exactly to the loading curve. To ensure that the loading curves are bounding in terms of maximum neutron multiplication factor, reactor record uncertainty and assembly operating history were also considered as discussed below. In reality, loaded casks will not be as reactive as the analyses predict due to the conservative modeling representations used and the fact that the large majority of assemblies loaded will have burnups greater than the minimum required burnup for loading (i.e., not all assemblies being loaded will be exactly on the loading curve).

3.1 Reactor Record Uncertainty

Reviews of the accuracy of reactor record–assigned burnup values for commercial SNF assemblies indicate that the uncertainty in these values is less than 5% [21, 22]. To account for reactor record uncertainty, a 5% burnup adjustment was made to the criticality loading curve (i.e., increasing the minimum burnup requirement by 5%). Note that although the methods used to calculate and verify assembly burnup values are documented in procedure form in NRC-approved technical specifications, these methods and the record-keeping methods of nuclear utilities are not uniform. In a few cases, some older SNF assemblies may have assigned burnup values that were averages for a batch of assemblies with similar characteristics. In such cases, an additional step will be required to convert the batch-average burnup value to assembly-specific burnup values prior to the assemblies being considered for loading into a waste package.

3.2 Bounding Spent Fuel Isotopic Compositions

The operating history of a fuel assembly as well as the fuel assembly design and effects of the neutron spectrum during irradiation can have a significant effect (several percent in reactivity) on end-of-life residual reactivity. Since the detailed operating history for each fuel assembly to be disposed of in the repository is not readily available for use in licensing analyses, a conservative approach must be used to generate limiting isotopic compositions (with respect to criticality calculations). Hence, when using a depletion code in burnup credit applications for commercial spent nuclear fuel (CSNF), operating history parameters must be selected to maximize fissile isotope production (harden neutron spectrum) in order to ensure that the calculated isotopic compositions for a given initial enrichment and burnup will yield a higher k_{eff} value than any assembly (with similar initial enrichment and burnup) depleted in a reactor. Therefore, in addition to using the design-basis configurations for generating the loading curves, which provide a conservative (with respect to criticality) geometric representation of the system, material compositions that increase system reactivity were also used.

To select conservative depletion parameters, a series of depletion parameter sensitivity evaluations was performed to determine relevant effects on end-of-life residual reactivity. Confirmation of conservatism for selected depletion parameters was performed using RCA-measured isotopic concentrations compared with calculated isotopic compositions, as well as comparisons using fuel assembly core following calculated isotopic concentrations with conservative depletion parameter–calculated isotopic compositions [23, 24, 25]. The depletion parameters selected to provide conservative spent fuel isotopic concentrations are provided in Table 3 along with typical nominal values provided in brackets. Plots showing the difference in k_{eff} between using bounding depletion parameters versus nominal depletion parameters for PWR and BWR assemblies in a waste package configuration are provided in Fig. 3 and Fig. 4, respectively, which show the amount of conservatism provided by using bounding depletion parameters.

The combined use of the bounding conditions and parameters in the depletion calculations and the application of the Δk_{ISO} term in the determination of the critical limit provides assurance that the isotopic compositions for the commercial SNF are handled in a conservative manner.

Table 3. Selected Conservative Depletion Parameters

Parameter	PWR	BWR
Assembly design	B&W 15 × 15	GE 7 × 7
Fuel temperature (K)	1144.1 [861.3]	1200 [1000]
Moderator temperature (K)	588.7 [579.8]	560.7 [560.7]
Moderator density (g/cm^3)	0.6905 (0.7556)	0.3 [0.43 length avg.)
Soluble boron concentration (ppmB)	1000 ppmB (constant) [letdown curve per cycle]	N/A
Burnable poison rods (B_4C for PWRs) Gd_2O_3 fuel rods for BWRs	Inserted in all tubes for all cycles even if depleted (3.5 wt % B_4C)	Not modeled[a] [varies per assembly]
Control blades	N/A	Inserted for final 15 GWd/MTU of irradiation
Fuel density (98% theoretical density) (g/cm^3)	10.741 [10.121 vol. avg.]	10.741 [≤10.4]
Specific power (MW$_t$/MTU)	29.74 [43.0 varies]	22.38 [35.68 varies]

[a]The BWR sensitivity analysis concluded that control blade insertion has the greatest impact on reactivity of any of the depletion parameters analyzed. Use of a full-length insertion over an extraordinarily long depletion period is bounding for all anticipated reactor operation scenarios and negates the need to model burnable absorbers ([25], Section 5.1.1.5).

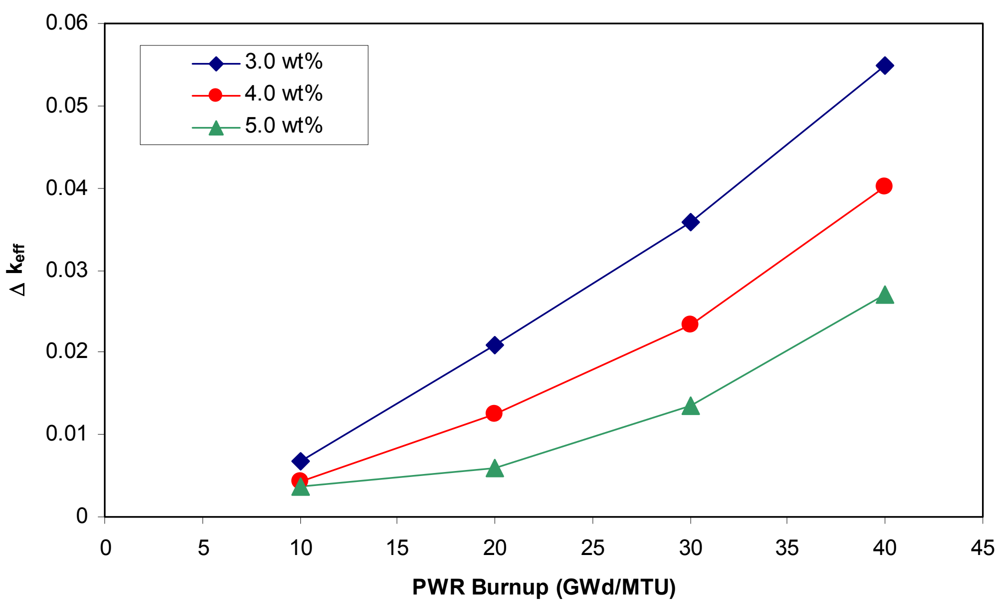

Figure 3. PWR Δk_{eff} Values for Bounding versus Nominal Depletion Parameters.

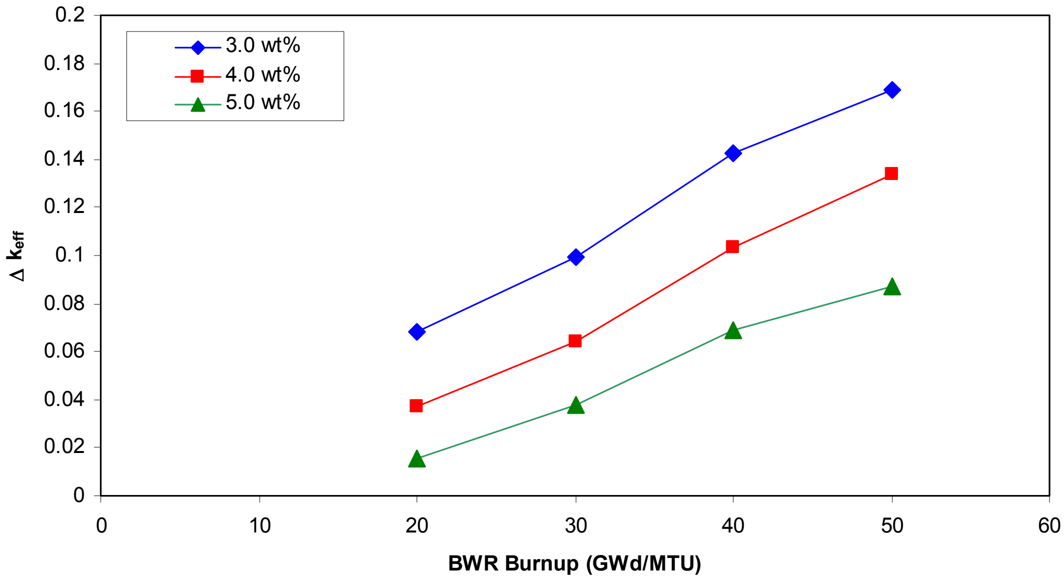

Figure 4. BWR Δk_{eff} Values for Bounding versus Nominal Depletion Parameters.

4. Misload Analysis

Loading casks in accordance with a burnup credit loading curve, which delineates acceptable and unacceptable fuel assemblies for loading, presents opportunities for misloading of assemblies that must be factored into the criticality evaluation. Misloading of an underburned fuel assembly causes an increase in reactivity. The extent of the increase is dependent on several factors but is dominated by the amount by which the actual assembly burnup is less than the minimum burnup value for loading acceptance and the position of the assembly within the cask.

Interim Staff Guidance on burnup credit (ISG-8)[26] for spent fuel in storage and transportation casks, issued by the NRC's Spent Fuel Project Office, recommends a burnup measurement for each assembly to confirm the reactor record and compliance with the assembly burnup value used for loading acceptance. This recommendation is intended to prevent unauthorized loading (misloading) of assemblies due to inaccuracies in reactor burnup records and/or improper assembly identification. The licensing basis was varied from this recommendation because the probability of criticality conditional upon a misload was incorporated into the overall determination of the total probability of criticality in the repository. Whether measurements are made or not does not eliminate the human error probability. The conservative approaches used to develop and apply the criticality loading curve and establish the probability of criticality conditional upon a misload were considered sufficiently robust that the utility-assigned burnup would be an adequate source of burnup values and additional means of verification of assigned burnup through physical measurements would not be needed.

The probability of exceeding the critical limit as a result of loading a *single* fuel assembly with insufficient burnup was evaluated for the licensing basis in *CSNF Loading Curve Sensitivity Analysis* [20]. Only a single misloaded assembly per package was evaluated based on a human factor error analysis that indicated that the likelihood of misloading more than one fuel assembly per package was below the probability threshold for consideration ([27], p. 22). Note that the actual processes for loading operations and confirmation were not defined, so certain assumptions in the analyses would require confirmation or a demonstration that they are bounding.

The misload analysis is based on the waste stream that corresponds to CSNF assemblies discharged from U.S. reactors through the end of 1999. This waste stream data did not indicate any BWR fuel assemblies on the unacceptable side of the loading curve (see Fig. 2), thereby obviating the need for a BWR cask misload evaluation. However, in the event that BWR assemblies were to be identified on the unacceptable side of the loading curve, the probability of criticality from this situation would be expected to be much less than that calculated for PWR waste packages.

A discretized representation of the PWR fuel inventory is shown in Fig. 2. The assemblies falling below the indicated PWR loading curve are potential misload assemblies and comprise the "misload inventory." Those falling above the curve are acceptable assemblies and are referred to as compensating assemblies and comprise the "compensating inventory." Of the 93,770 assemblies in the inventory, a total of 1,990 (2.1%) were viewed as potential misload assemblies.

Because of symmetry, there would be five unique basket positions in the 21-PWR basket that would yield different reactivity changes if loaded with an underburned fuel assembly. These basket positions are denoted with letter designations in Fig. 5. For the licensing basis, a distinct misload analysis was performed for each position, which evaluated the probability that a misload in that position leads to a k_{eff} value exceeding the critical limit. The combined failure probability was then determined by combining the individual position results using the position multiplicities (the number of symmetrically identical positions of each type in the basket) as a weight function. If $P_{f,i}$ is the failure probability for a misload in position i with multiplicity M_i, then the failure probability, P_f, is as follows

$$P_f = \frac{\sum_i M_i P_{f,i}}{\sum_i M_i} \qquad \text{(Eq. 4)}$$

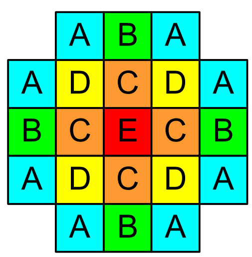

Figure 5. Waste Package Cell Location Idenitifiers.

The failure probabilities for each unique basket position, $P_{f,i}$, were determined by a stochastic analysis. In this analysis, waste packages were loaded based on random samples from the spent fuel inventory and the waste package reactivity was calculated and compared with the critical limit. For each misload position, a single assembly was uniformly sampled from the misload inventory. The remaining basket positions were filled by uniform sampling from the compensating assembly inventory (where a maximum burnup of 50 GWd/MTU was credited). The samples with reactivities greater than the critical limit were tallied, and an estimate of the failure rate was obtained from the maximum likelihood estimator for a binomial distribution

$$P = \frac{k}{n} \qquad \text{(Eq. 5)}$$

where k is the number of failures in n total trials.

The combined failure probability for all misload positions was calculated to be 0.2% and presented by location in Table 4. The licensing basis conservatively used the failure probability for the worst case "E" position of 1.4% in calculations for the probability of one or more criticalities in the repository. It should be noted that misload positions "A" and "B" on the periphery of the basket did not result in any failures (i.e., cases that exceeded the critical limit). It should also be noted that based on the assembly inventory, there were only 36 assemblies that resulted in exceeding the critical limit ([20], p. 6-91).

Table 4. Combined Failure Probability for Each Misload Position

Misload location	Multiplicity	P fail (%)	σ (%)
A	8	0.0	0.00
B	4	0.0	0.00
C	4	0.8	0.03
D	4	0.1	0.00
E	1	1.4	0.03
Total		0.2	0.01

Although the combination of RCAs and critical experiments has covered all of the 29 principal isotopes, some are limited in number or come from a single source. Therefore, a series of sensitivity evaluations were performed with reduced sets of spent fuel isotopes to assess the impacts on the loading curve and resultant number of unacceptable assemblies. The results are illustrated in Fig. 6 [20]. The legend is defined as follows: (a) "Principal Actinide" (PA) corresponds to the actinide isotopes listed in Table 1. (b) "Actinide Only" sets correspond to the "Principal Actinde" set minus ^{237}Np, ^{242m}Am, and ^{243}Am. (c) The "Metal" isotope subset is comprised of ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, and ^{109}Ag. By excluding this subset from the Principal Isotope set, the "PI—Metal" subset is defined. (d) The "French" subset is comprised of ^{103}Rh, ^{143}Nd, ^{149}Sm, ^{152}Sm, and ^{155}Gd. This subset combined with the

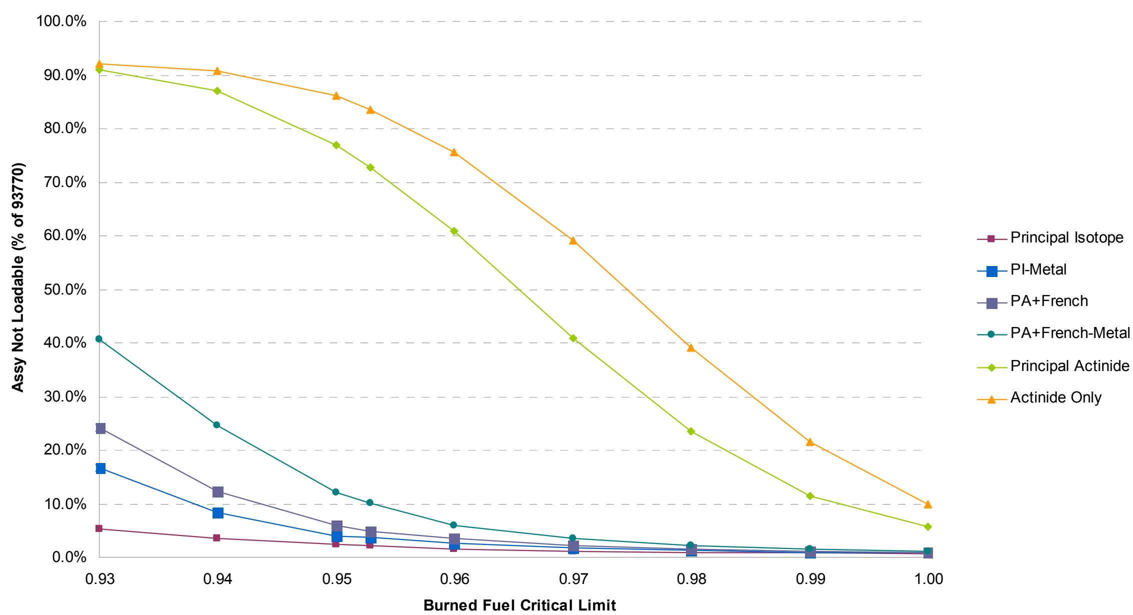

Figure 6. Loading Curve Sensitivity to Credited Isotope Set. *Source:* C. S. Henkel, *CSNF Loading Curve Sensitivity Analysis* [20].

Principal Actinide set defines the “PA + French” isotope set. By further excluding the “Metal” subset from this set, the “PA + French—Metal” isotope set is defined. The results show that as the number of isotopes represented in the spent fuel composition increases, the loading curve becomes less sensitive to changes in the critical limit. Note that the critical limit moves to the right as the number of credited isotopes is decreased.

5. Summary

This paper provides a description of the burnup credit implementation in the Yucca Mountain license application. In applications where burnup credit is requested, it is important to recognize how the results of burnup credit will be used to determine acceptance criteria. DOE used burnup credit in criticality evaluations with a risk-informed process by performing a probabilistic assessment in accordance with 10 CFR 63 to determine whether the consequences of a critical event should be factored into the TSPA. Ultimately, crediting the reduced reactivity potential of irradiated fuel (burnup credit) is a necessary component for demonstrating criticality prevention during postclosure.

6. Acknowledgment

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