REVIEW OF YUCCA MOUNTAIN DISPOSAL CRITICALITY STUDIES*

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The U.S. Department of Energy (DOE), Office of Civilian Radioactive Waste Management, submitted a license application for construction authorization of a deep geologic repository at Yucca Mountain, Nevada, in June 2008. The license application is currently under review by the U.S. Nuclear Regulatory Commission. However, on March 3, 2010, DOE filed a motion requesting withdrawal of the license application. With the withdrawal request and the development of the Blue Ribbon Commission to seek alternative strategies for disposing of spent fuel, the status of the proposed repository at Yucca Mountain is uncertain. What is certain is that spent nuclear fuel (SNF) will continue to be generated and some long-lived components of the SNF will eventually need a disposition path(s). Strategies for the back end of the fuel cycle will continue to be developed and need to include the insights from the experience gained during development of the Yucca Mountain license application. Detailed studies were performed and considerable progress was made in many key areas in terms of increased understanding of relevant phenomena and issues regarding geologic disposal of SNF. This paper reviews selected technical studies performed in support of the disposal criticality analysis licensing basis and the use of burnup credit. Topics include assembly misload analysis, isotopic and criticality validation, commercial reactor critical analyses, loading curves, alternative waste package and criticality control studies, radial burnup data and effects, and implementation of a conservative application model in the probabilistic criticality evaluation—as well as other information that is applicable to operations regarding spent fuel outside the reactor. This paper summarizes the work and significant accomplishments in these areas and provides a resource for future, related activities.

I. INTRODUCTION

Expected outcomes of the Department of Energy's (DOE's) request to withdraw the Yucca Mountain (YM) license application (LA) are an increased reliance on long-term storage of spent nuclear fuel (SNF) and a reevaluation of options for the back end of the fuel cycle. To provide flexibility operationally as well as economically, there continues to be considerable interest in the United States (U.S.), as well as internationally, in the increased use of burnup credit (BUC), particularly related to storage, transport, reprocessing, and disposal of commercial SNF. For YM, BUC was used only for evaluations of in-package configurations and used a combination of conservative and bounding modeling approximations. A summary overview of the BUC implementation for the YM LA is available in *Burnup Credit Approach Used in the Yucca Mountain License Application.*[1]

The postclosure nuclear criticality analysis methodology followed a risk-informed, performance-based process presented in *Disposal Criticality Analysis Methodology Topical Report.*[2] The initial version of this report was accepted by the U.S. Nuclear Regulatory Commission (NRC) with the issuance of a Safety Evaluation Report.[3] The methodology included taking credit for the reduced reactivity potential of irradiated commercial light-water reactor (LWR) fuel assemblies in criticality analyses (i.e., BUC). Computational biases and uncertainty were established based on a combination of publicly available information and proprietary data, including commercial reactor criticals (CRCs), laboratory critical experiments (LCEs), and destructive radiochemical assay (RCA) data. Destructive RCA data were used to determine bias in isotopic concentration predictions, and CRCs and LCEs were used to determine bias in k_{eff} predictions.

Consistent with the governing regulations in 10 CFR 63, the strategy for licensing the proposed repository at YM was based on a total performance assessment (PA). A PA is an analysis that identifies features, events, and processes (FEPs) and sequences of events and processes that might affect the total disposal system, including both natural and engineered systems, and their probabilities of occurring. Within the probabilistic approach, criticality was considered an event, and the total probability of a criticality event during the disposal time period was calculated and compared against the regulatory criterion in 10 CFR 63. The total probability of criticality includes contributions associated with both internal (within the waste package) and external (external to the waste package) criticality. The probability of criticality was estimated based on a combination of low probability events that were YM site-specific and a fixed disposal inventory. Criticality was screened from

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consideration in the PA 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."

Numerous studies were performed over the years in support of the YM LA, and the primary objective of this paper is to bring awareness to these studies that have relevance to future efforts related to criticality of SNF in storage, transport, reprocessing, and disposal. Permanent geological isolation will be required for at least some long-lived components of SNF regardless of fuel cycle strategy. The following sections provide a brief review of selected disposal criticality technical studies.

II. REVIEW OF TECHNICAL STUDIES

II.A. Spent Fuel Isotopic Compositions

The validation of the isotopic calculations considered CRC and 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. The bias and bias uncertainty of the CRC and RCA data k_{eff} values were evaluated in *Isotopic Model for Commercial SNF Burnup Credit*[4] based on a direct difference approach.[5] The RCA data results bounded the CRC data results; therefore, it was selected for use in the YM LA. For the RCA data, the combined bias and bias uncertainty was established by comparing criticality calculations performed using measured isotopic concentrations from 104 (74 PWR, 30 BWR) assay samples with criticality calculations performed using calculated isotopic concentrations for the assay samples. The one-dimensional (1-D) SAS2H depletion code was used for calculating the isotopic concentrations. The combined bias and bias uncertainty based on the measured RCA data was calculated to be −0.0249 Δk using a single-sided lower tolerance limit established such that there is a 95% confidence that at least 95% of the population is above the limit.[4] Note that the large Δk is primarily a result of uncertainties in the RCA data.

II.A.1. SCALE 5.1 Predictions of PWR Spent Nuclear Fuel Isotopic Compositions

Subsequent to the submittal of the YM LA, the isotopic concentration calculations for the PWR assay samples were redone[6] with additional RCA data and the two-dimensional (2-D) depletion capabilities in SCALE 5.1 (Ref. 7). Specifically, the 2-D transport and depletion module, TRITON/NEWT, and the 44GROUPNDF5 cross-section library in the SCALE 5.1 code system were used.

The calculation report describes a comprehensive evaluation of RCA measurement, assembly design, and irradiation history data for 118 PWR spent fuel samples obtained from low-, moderate-, and high-burnup spent fuel assemblies from nine PWRs: Trino Vercellese, Kernkraftwerk Obrigheim (KWO), Turkey Point Unit 3, H. B. Robinson Unit 2, Calvert Cliffs Unit 1, Three Mile Island (TMI) Unit 1, Takahama Unit 3, Gösgen, and GKN II. The initial fuel enrichments and burnup values for the samples considered vary from 2.453 to 4.657 wt % ^{235}U and from 7.2 to 70.4 GWd/MTU, respectively. The physical parameters and irradiation conditions of the spent fuel samples are representative of the range of initial enrichment and burnup values for the inventory of commercial spent fuel assemblies in the U.S. The sample range is illustrated in Fig. 1 superimposed over a generic loading curve and representative PWR spent fuel inventory.

The YM LA BUC methodology was based on a selected set of actinides and fission products referred to as the *Principal Isotopes*.[2] Within the majority of the RCA data sets, the number of measured nuclides is substantially greater than the list of principal isotopes considered in burnup credit for criticality safety evaluations. The additional isotopes include many that are of radiological importance (neutron and gamma ray emitters, e.g., ^{244}Cm and ^{137}Cs), are important to decay heat (e.g., ^{134}Cs), and others that are important to longer term waste management safety analyses (e.g., ^{79}Se, ^{135}Cs). The intent of this report was not only to serve as a radiochemical analysis database for burnup credit isotopic validation, but also validation for many other safety- and design-related applications important to out-of-reactor analyses. The complete list of nuclides evaluated is listed below with the YM principal isotopes shown in italics. Note that ^{233}U is a YM principal isotope but is not present in spent fuel in appreciable quantities until thousands of years due to decay of ^{237}Np.

Actinides
- *^{234}U*, *^{235}U*, *^{236}U*, *^{238}U*
- *^{238}Pu*, *^{239}Pu*, *^{240}Pu*, *^{241}Pu*, *^{242}Pu*
- *^{237}Np*, *^{241}Am*, *^{242m}Am*, *^{243}Am*
- ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm

Lanthanides
- *^{143}Nd*, ^{144}Nd, *^{145}Nd*, ^{146}Nd, ^{148}Nd, ^{150}Nd
- ^{144}Ce, ^{147}Pm
- *^{147}Sm*, ^{148}Sm, *^{149}Sm*, *^{150}Sm*, *^{151}Sm*, *^{152}Sm*, ^{154}Sm
- *^{151}Eu*, ^{152}Eu, *^{153}Eu*, ^{154}Eu, ^{155}Eu
- ^{154}Gd, *^{155}Gd*, ^{156}Gd, ^{157}Gd, ^{158}Gd, ^{160}Gd

Volatile fission products
- ^{133}Cs, ^{134}Cs, ^{135}Cs, ^{137}Cs

*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.

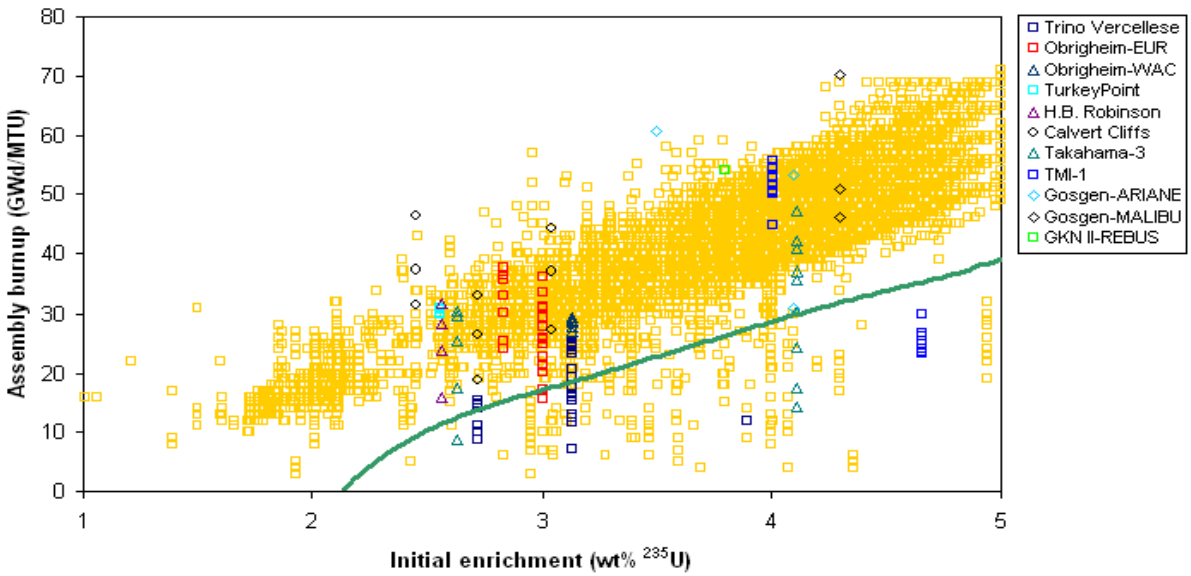

Fig. 1. Enrichment and burnup values of spent fuel samples used for validation compared to the actual and projected spent fuel assembly inventory and typical loading curve for PWR SNF assuming actinide and fission product burnup credit. (Note: Each point for the existing inventory represents multiple assemblies.)

Metallic and other fission products
- ^{90}Sr, ^{95}*Mo*, ^{99}*Tc*, ^{101}*Ru*, ^{106}Ru, ^{103}*Rh*, ^{109}*Ag*
- ^{79}Se, ^{105}Pd, ^{108}Pd, ^{126}Sn, ^{129}Sb

To assess the impact of the updated isotopic concentration calculations in terms of Δk, *Propagation of Isotopic Bias and Uncertainty to Criticality Safety Analyses of PWR Waste Packages* was developed.[8] This report determines the impact on the calculated effective neutron multiplication factor, k_{eff}, due to the bias and bias uncertainty associated with predicted spent fuel compositions for use in criticality analyses employing burnup credit.

A stochastic approach[9] was used for estimating the total uncertainty in k_{eff} resulting from uncertainties in the calculated isotopic compositions based on Monte Carlo sampling of probability distributions obtained for the isotopic bias. This method simulates the variation of bias within the range of bias uncertainty by randomly sampling the uncertainty values. The sampling distributions established for the measured-to-calculated isotopic concentration values are assumed to be normal with tolerance intervals used to define the range of bias uncertainty. The randomly sampled bias uncertainties for the principal isotopes are applied to the predicted fuel concentrations to determine new fuel isotopic concentrations for use in criticality calculations. A statistically significant number of criticality calculations is required to ensure that the sampled bias uncertainty values are representative of the underlying distribution of the isotopic variance.

Calculated results using this methodology, the additional RCA data, and updated isotopic concentration calculations resulted in a combined isotopic composition bias and bias uncertainty in terms of Δk_{eff} of -0.0229, as compared to the value of -0.0249 used in the YM LA.

II.A.2. SCALE 5.1 Predictions of BWR Spent Nuclear Fuel Isotopic Compositions

This report[10] was developed subsequent to the YM LA submittal with a purpose of documenting the comparison of measured isotopic concentrations for BWR SNF to calculated concentrations determined with SCALE 5.1. The calculation methodology employed the 2-D transport and depletion module, TRITON/NEWT, and the 44GROUPNDF5 cross-section library.

The calculation report describes an evaluation of RCA measurement, assembly design, and irradiation history data for 32 BWR spent fuel samples obtained from fuel assemblies from three BWRs: Fukushima Daini Unit 2 (Fukushima Daini-2), Cooper, and Gundremmingen-A. The initial fuel enrichments for the samples varied from 2.53 to 3.91 wt % ^{235}U, and burnup values ranged from 14.39 to 43.99 GWd/MTU. Measurements for 40 isotopes were evaluated in this study with a summary of average calculated-to-measured (C/M) ratios provided in Table I. Due to termination of the YM project these values were not propagated to bias and bias uncertainty impacts on k_{eff} for BWR waste packages.

TABLE I. Summary of BWR Isotopic Inventory Ratios[a]

Nuclide	Average C/M	Standard Deviation	Number of Samples
^{234}U	1.034	0.042	20
^{235}U	0.995	0.054	32
^{236}U	0.980	0.021	32
^{238}U	0.999	0.005	32
^{237}Np	0.997	0.115	20
^{238}Pu	0.950	0.087	32
^{239}Pu	1.012	0.045	32
^{240}Pu	1.000	0.035	32
^{241}Pu	0.988	0.055	32
^{242}Pu	1.012	0.062	32
^{241}Am	1.087	0.121	20
^{242m}Am	1.133	0.122	14
^{243}Am	1.130	0.067	14
^{242}Cm	0.590	0.179	26
^{243}Cm	0.820	0.065	14
^{244}Cm	0.964	0.120	32
^{245}Cm	0.626	0.095	14
^{246}Cm	0.548	0.178	13
^{143}Nd	1.003	0.014	14
^{144}Nd	0.996	0.081	14
^{145}Nd	1.013	0.012	14
^{146}Nd	1.005	0.014	14
^{148}Nd	0.992	0.013	26
^{150}Nd	1.009	0.011	14
^{137}Cs	0.975	0.060	32
^{135}Cs	1.054	0.037	6
^{134}Cs	0.893	0.101	24
^{154}Eu	0.890	0.072	24
^{144}Ce	0.940	0.125	13
^{147}Sm	0.926	0.039	11
^{148}Sm	0.883	0.046	11
^{149}Sm	1.039	0.127	11
^{150}Sm	0.989	0.026	11
^{151}Sm	1.229	0.048	11
^{152}Sm	1.157	0.063	11
^{154}Sm	0.931	0.033	11
^{79}Se	1.352	0.030	6
^{90}Sr	1.075	0.017	6
^{99}Tc	1.146	0.036	6
^{126}Sn	2.997	0.072	6

[a]Principal isotopes italicized.

II.B. Criticality Application Model

The majority of the YM criticality analyses used the general-purpose Monte Carlo N-Particle (MCNP) transport code[11] to analyze a wide variety of potential configurations and modeling assumptions. A key element in all criticality calculations is the geometric configuration of the fissile material. The commercial SNF disposal strategy for YM included using transportation, aging, and disposal (TAD) canisters to transfer the majority of the waste from the generating sites to the geologic repository operations area. DOE-owned SNF transport and disposal was also based on a standardized canister concept. Since long-term geologic storage and/or disposal must consider other parameters affecting the repository that are temporally and spatially dependent, a number of disposal canister internal configurations are possible over time. As the repository, waste package, and TAD designs for YM evolved, comprehensive criticality evaluations were performed to determine the configurations and parameters of influence that yielded the highest system k_{eff} values after waste emplacement and during different stages of degradation. These evaluations led to the development of design-basis models[12] 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 TAD and SNF loading criteria were developed such that under fully flooded conditions and allowances for corrosion, subcriticality would be maintained. The design basis model was developed using the most reactive fuel assembly design in terms of highest residual reactivity. Using the most-reactive assembly design simplified the YM LA by providing a bounding value for comparison to other designs. Sensitivity evaluations were performed[13] and identified the B&W 15×15 and GE 7×7 as bounding for the TAD configurations. Neutron absorber plates were credited with 75% of design-specified absorber material, consistent with existing NRC guidance[14] for fixed absorbers in dry cask storage, and reduced in thickness to account for 10,000 years of general corrosion. Spent fuel isotopic compositions were represented with a 5-year cooling time and calculated using depletion model parameters that increase residual reactivity at discharge. Sensitivity evaluations were performed for parameter selection, and databases of bounding isotopic compositions were generated.[15,16,17] Selected depletion parameters are presented in Table II.

TABLE II. Conservative Depletion Parameters

Parameter[a]	PWR	BWR
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	1000 ppm B [letdown/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[b] [varies per assembly]
Control blades	N/A	Inserted for final 15 GWd/MTU of irradiation

TABLE II (continued)

Parameter[a]	PWR	BWR
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]Bracketed [] values show typical nominal parameters.

[b]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.[17]

II.B.1. Radial and Axial Profile Effects

As fuel is burned in a reactor, the burnup of the fuel becomes distributed axially and radially as the reactivity of the fuel decreases. The axial burnup distribution attains a flattened cosine shape with time, although the exact profile will vary with operating history and other effects unique to the individual reactor. The cosine shape is representative of typical burnup profiles,[18] which shows that the ends of the fuel are less burned than the central region. The impact of the lower burned ends can have a significant impact on the system k_{eff} for a given burnup.[19] Several variations in axial profiles were evaluated in *PWR Axial Burnup Profile Analysis*[20] for use in developing loading curves with the criticality application model. Conservative profiles were developed based on using statistical tolerance limits.

Radial variations in the neutron flux in a reactor core, which are mainly due to leakage at the core periphery and reactivity control components, result in a nonuniform horizontal burnup distribution over the radial extent of the reactor core. As a reactor operates, the radial flux shape flattens due to fuel depletion and fission product poisoning near the core center. However, because of the high leakage, burnup drops off rapidly near the core periphery. At the end of a cycle, the individual assemblies located near the center of the core will have a relatively uniform horizontal burnup distribution, while the assemblies near the core periphery may have a significant horizontal variation in burnup.[21] Thus it is possible for fuel rods on one side of an assembly to have experienced less burnup than fuel rods on the opposite side of the same assembly. Typically this effect is flattened through assembly shuffling when core reloads are performed. However, horizontal burnup variation could result in an increase in reactivity as compared to uniform horizontal burnup—hence only a criticality concern if not properly treated.

PWR Radial Burnup Gradient Reactivity Evaluation[22] evaluated the effect of radial burnup (gradient across fuel assembly) on the reactivity of commercial SNF loaded in TAD canisters to develop a technically justifiable approach for addressing radial burnup gradients in the postclosure criticality analysis. The analysis and results were based on radial burnup data from a representative set of PWR assemblies.[21] Conclusions of this work indicated that the influence of the radial burnup gradient on the system k_{eff} was dominated by (1) the magnitude of the radial burnup gradient and (2) the orientation of the assemblies within the waste package. The results indicated that when the axial burnup is included in the computational analysis, the effect of radial burnup gradient does not exceed ~0.9% $\Delta k/k$ (value based on both the worst-case loading configuration and bounding radial burnup gradients). The calculations also showed that reductions in the radial burnup gradients notably reduced the effects. In summary, the analysis indicated that the increase in k_{eff} of a waste package due to actual assembly radial burnup gradients is expected to be very small, as compared to k_{eff} values calculated based on uniform radial burnup. This conclusion is specific to a given design basis configuration used for generating assembly loading specifications. If loading specifications are generated using a different design-basis representation, then the results and conclusions need to be reevaluated. Future work in this area could use the methodology from this report in conjunction with a more recent, larger database of radial burnup profiles,[23] which also includes BWR data.

In *YMP Postclosure Support—Radial Burnup Gradient and Fuel Characteristics Database—Non-Proprietary*,[23] two databases of radial burnup gradient data and reactor operating history data are provided. One is for PWR fuel assemblies and the other is for BWR fuel assemblies. The data for both PWRs and BWRs were generated by AREVA fuel depletion codes. In total, 3304 node-burnup pairs were provided from four axial nodes of 826 PWR assemblies. For the BWRs, four axial nodes of 659 assemblies were analyzed, producing 2636 data points. The reactor-specific data is summarized in Tables III and IV for the PWR and BWR assemblies, respectively.

TABLE III. Collected PWR Assembly Data

Reactor Type	Assembly Type	Number of Assemblies	Number of Node/Burnup Pairs
CE-213	CE 14×14	147	588
CE-213	CE 14×14	111	444
W-157	W 17×17	79	316
W-157	W 17×17	85	340
B&W 15×15	B&W 15×15	87	348
B&W 15×15	B&W 15×15	71	284
W-193	W 17×17	83	332
W-193	W 17×17	82	328
B&W-177	B&W 15×15	81	324

TABLE IV. Collected BWR Assembly Data

Reactor Type	Assembly Type	Number of Assemblies	Number of Node/Burnup Pairs
BWR-4	GE13/ GE14	149 / 124	596 / 496
BWR-4	ATRIUM-10	124	496
BWR-5	SVEA96	120	480
BWR-6	GE11	142	568

II.C. Burnup Credit Loading Curves

The YM LA criticality loading curves[12] were established such that the k_{eff} of a waste package fully loaded with assemblies selected from the curve would be less than the critical limit under postulated postclosure conditions, including igneous scenarios.[24] The k_{eff} limit used was an actual critical limit (CL), as opposed to an upper subcritical limit, because the purpose of the disposal criticality evaluation was to determine the probability of a criticality event over the regulatory specified initial FEP screening time period (i.e., 10,000 years) to establish whether it should be included in the PA. Traditional criticality safety is typically based on a deterministic assessment where an additional administrative margin is applied to the CL to ensure subcriticality. The criticality loading curves were used in probabilistic evaluations to calculate the probability of criticality as a result of a misload (i.e., inappropriate assembly loading or inappropriate basket loading). Hence, the initial package loading process requires stringent controls to ensure that probability objectives for disposal are met.

The YM licensing basis was developed to accommodate multiple canister criticality control design configurations because licensed TAD canister designs did not exist. This was accomplished by using the design-basis application model discussed above. It is recognized that this can result in significant conservatism for reactor sites that have assembly designs different from the design basis. However, using the design basis for all TADs enabled a bounding probability of criticality[25] occurrence in the repository to be calculated for the LA.

A number of internal packaging design variants were considered before finally selecting what was used in the YM LA that may provide insight for other spent fuel handling and disposal considerations. Some early designs were developed under the premise that fuel would be loaded into packages under dry conditions, allowing the use of basket materials that may not be appropriate for wet handling operations (e.g., carbon steel). With renewed emphasis on long-term storage and the realization that some number of existing storage casks may require repackaging of fuel at a later time, dry-transfer fuel systems may be considered. Some of the earlier design studies[26] showed that criticality could be controlled by moderator displacement from corrosion products. Other studies[27] showed that because BWR assemblies are channeled, they inherently create regions for increased neutron thermalization acting as flux traps between absorber plates when flooded. The thermalization region decreases when materials corrode and displace moderator in a BWR package, hence diminishing the neutron absorber effectiveness.

II.C.1 Critical Limit

BUC loading curves specify loading criteria in terms of the minimum required assembly burnup as a function of assembly-initial enrichment. A loading curve represents combinations of burnup and initial fuel enrichment that correspond to a single value of k_{eff} (i.e., critical limit) for a given configuration. 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 (Ref. 28).

The CL for the YM LA was calculated using Equation 1.

$$CL(x) = f(x) - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m \,, \qquad (1)$$

where

x = a neutronic parameter used for trending

$f(x)$ = the lower-bound tolerance limit (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 (-0.0259 discussed above)

Δk_m = traditional administrative margin to ensure subcriticality (0.0 for probability of criticality evaluations).

The commercial SNF LBTL was developed[29] using critical experiments selected based on neutronic similarity with the design-basis system. Neutronic similarity was determined using the sensitivity/uncertainty (S/U) analysis methods and tools in SCALE. 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*,[30] proprietary HTC mixed-oxide critical experiments,[31] and CRCs.[32] 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.

The LBTL was established for MCNP with nuclear data based primarily on the ENDF/B-VI library. Only critical benchmarks applicable to the design basis configuration (totaling ~240 experiments) were used for calculating the LBTL at a 95% confidence level for 95% of the population. The LBTL and corresponding range of applicability was determined to be 0.9905 for fresh fuel waste packages with an energy of the average lethargy of a neutron causing fission (*EALF)* range of 0.0977 to 0.3882 eV; 0.9778 for waste packages containing 21 burned PWR fuel assemblies over an EALF range of 0.0684 to 1.0410 eV; and 0.9778 for waste packages containing 44 burned BWR fuel assemblies over an EALF range of 0.0421 to 0.9679 eV.

Additional uncertainties were considered such as reactor record burnup uncertainty and half-life and branching fraction uncertainty.[33] Reviews[34,35] of the accuracy of reactor record assigned burnup values for commercial SNF assemblies indicate that the uncertainty in these values is less than 5%. Therefore, a conservative uncertainty value of 5% was used to adjust 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. Some SNF assemblies may have assigned burnup values that are averages for a batch of assemblies with similar characteristics. In such cases, an additional step was 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. The assigned burnup value of each assembly at the time of loading is important information that needs to be tracked until the fuel is permanently dispositioned.

Half-life and branching fraction uncertainties were quantified in terms of their effects on the neutron multiplication factor. Both the BWR and the PWR results showed that the reactivity uncertainty peaks around a decay of 50,000 years, drops to a minimum after a decay of 200,000 years, and increases slightly at a decay of 1,000,000 years. The results show that the principal contributors to the maximum uncertainty at a decay time of about 50,000 years are ^{239}Pu, ^{235}U, and ^{240}Pu. At a decay time of 1,000,000 years the principal contributors are ^{135}Cs, ^{237}Np, and ^{93}Zr. Test cases were analyzed for 10,000; 20,000; 30,000; 40,000; 50,000; 100,000; 200,000; and 1,000,000 year decay. Results indicated an uncertainty term of 3.75×10^{-4} $\Delta k/k$ would be bounding for assemblies with an enrichment up to 5.5 wt% ^{235}U and burnups up to 75 GWd/MTU for a decay period up to 1 million years.

II.C.2 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 system 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.

Traditional criticality safety analyses are performed in a deterministic manner and hence typically do not account for the probability of events, which can result in unrealistically bounding results. For the YM analysis, the fuel assembly misload analysis was done on a probability basis incorporating human factor error analyses.

Interim Staff Guidance on burnup credit (ISG-8)[36] 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 YM licensing basis deviated 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 probability of exceeding the critical limit as a result of loading a fuel assembly with insufficient burnup was evaluated for the YM licensing basis in *CSNF Loading Curve Sensitivity Analysis*.[12] Only a single misloaded assembly per package was evaluated[37] based on a human error analysis[38] which reviewed licensee event reports and other published media [e.g., Institute of Nuclear Power Operations (INPO) library database] that pertains to fuel-handling events and calculated the probability of fuel assembly damage and misload events using an event tree methodology. The human error analysis information indicated that the likelihood of misloading more than one fuel assembly per package was

below the probability threshold for consideration in 10 CFR 63. Note that the actual processes for loading operations and confirmation were not defined, so certain assumptions in the analyses would require confirmation or process development to ensure that they can be performed within a certain probability threshold to meet spent fuel disposition objectives.

The misload analysis was based on the waste stream that corresponds to commercial SNF 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, 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 in the future, the probability of criticality from this situation is expected to be much less than that calculated for PWR waste packages. A representative discretization of the PWR fuel inventory is shown in Fig. 1. Of the 93,770 assemblies in the inventory based on a certain arrival forecast[12] for the YM LA, 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. 2. A 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

$$P_f = \frac{\sum_i M_i P_{f,i}}{\sum_i M_i} \quad . \tag{2}$$

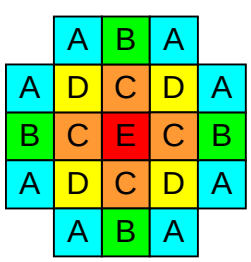

Fig. 2. Waste package cell location identifiers.

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 k_{eff} 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

$$P = \frac{k}{n} \quad , \tag{3}$$

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 is presented by location in Table V. 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, only 36 assemblies were calculated to exceed the critical limit.[12]

TABLE V. 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

Sensitivity evaluations were performed to assess the impacts of different levels of burnup credit (i.e., reduced sets of spent fuel isotopes) on the loading curve and resultant number of unacceptable assemblies. The results are illustrated in Fig. 3 (Ref. 12). The legend is defined as follows: (a) "Principal Actinide" corresponds to the actinide isotopes listed in italics in II.A.1 plus ^{233}U. (b) "Actinide Only" set corresponds to the "Principal Actinide" 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 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.

III. FUTURE CONSIDERATIONS

Burnup credit is typically associated with criticality safety but has broader implications when considering the longer storage periods and the ability to have flexibility in design to facilitate multiple disposition paths. It provides an accurate accounting of residual fissile and radionuclide content, allows process control within a reprocessing plant, and enables flexibility with thermal management and shielding strategies, not to mention the significant economic benefits associated with more accurate material characterization. It is important to recognize how the results of BUC can be used to determine appropriate acceptance criteria and how these can be properly integrated to meet storage, transportation, and disposal needs. Ultimately, burnup credit is a necessary component for all future strategies regarding closing the back end of the fuel cycle.

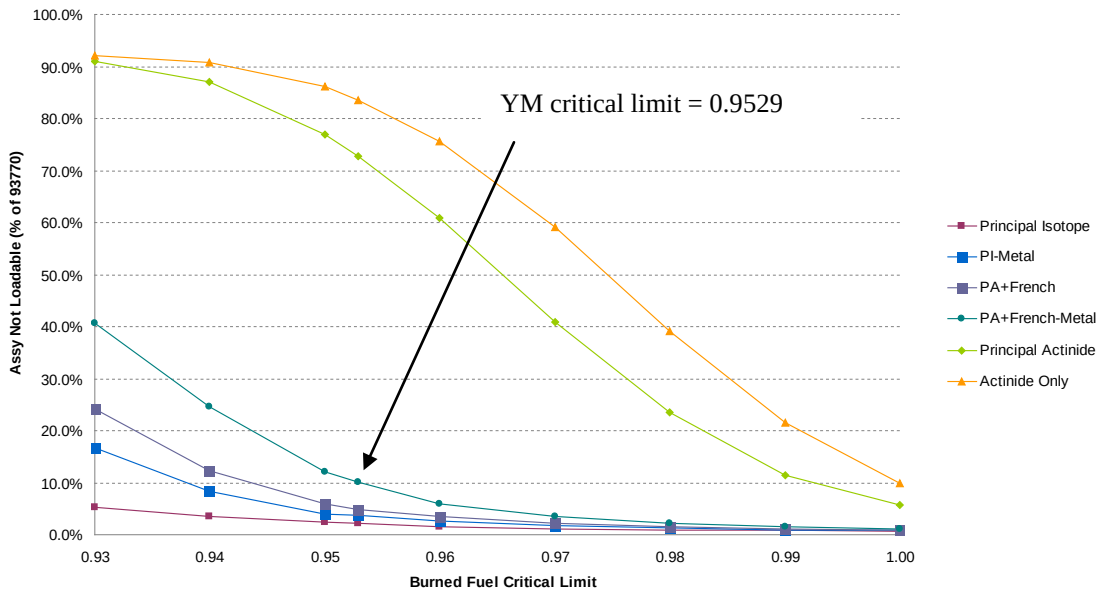

Fig. 3. Loading curve sensitivity to credited isotope set.

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