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Office of Civilian Radioactive Waste Management

Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages

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ABSTRACT

The Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages describes a methodology for performing and applying nuclear criticality safety calculations with actinide-only burnup credit. The changes in the U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241 concentration with burnup are used in burnup credit criticality analyses. No credit for fission product neutron absorbers is taken. The methodology consists of five major steps.

1. Validate a computer code system to calculate isotopic concentrations of spent nuclear fuel (SNF) created during burnup in the reactor core and subsequent decay. A set of chemical assay benchmarks is presented for this purpose, in conjunction with a method for assessing the calculational bias and uncertainty, and conservative correction factors are presented for each isotope.

2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff}, of a SNF package. UO_2 critical experiments needed for the fresh fuel assumption are agumented by 47 MOX critical experiments to cover the plutonium isotopes and Am-241. The method uses an upper safety limit on k_{eff} (which can be a function of the trending parameters) to assure that the calculated k_{eff}, when increased for the bias and uncertainty, is less than 0.95.

3. Establish bounding conditions for the isotopic concentration and criticality calculations. Three bounding axial profiles have been established to assure the "end effect" is accounted for conservatively.

4. Use the validated codes and bounding conditions to generate package loading criteria (burnup credit loading curves). Burnup credit loading curves show the minimum burnup required for a given initial enrichment. The utility's declared burnup is compared to this minimum burnup requirement after the utility accounts for the uncertainty in its burnup. Separate curves may be generated for each assembly design, various minimum cooling times, and burnable absorber histories.

5. Verify that SNF assemblies meet the package loading criteria and confirm proper assembly selection prior to loading. A measurement of the average assembly burnup is required.

Each step is described in detail for use with any computer code system and is then demonstrated with the SCALE 4.2 computer code package using 27BURNUPLIB cross sections.

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EXECUTIVE SUMMARY

Historically, safety analyses of criticality control systems for spent nuclear fuel (SNF) transportation packages include an assumption that the SNF loaded into the package is "fresh" or unirradiated. The "fresh fuel" assumption is very conservative since the potential reactivity of the nuclear fuel is substantially reduced after being irradiated in the reactor core. The concept of taking credit for this reduction in the reactivity of nuclear fuel due to burnup of the fuel, as opposed to using the fresh fuel assumption in the criticality safety analysis, is referred to as "Burnup Credit." Burnup credit uses the actual physical composition of the fuel and accounts for the net reduction of fissile material and the buildup of neutron absorbers in the fuel as it is irradiated. Using only the change in actinide isotopes in the burnup credit criticality analysis is referred to as "Actinide-Only Burnup Credit." Not using the fission products creates a margin that has been assigned to gain consensus where data is sparse.

Although the fresh fuel assumption represents a conservative design approach, which substantially simplifies the criticality safety analysis and associated administrative controls, it results in a significant reduction in SNF capacity for a given package weight. The use of burnup credit in the design of criticality control systems enables more spent fuel to be placed in a package. Increased package capacity in turn results in a reduced number of storage, shipping, and disposal containers for a given number of SNF assemblies. Fewer shipments result in a lower risk of accidents associated with the handling and transportation of spent fuel, thus reducing both radiological and non-radiological risk to the public. (Although there is a reduction in the radiological risk, this risk is already extremely small.) A reduced number of packages also reduces worker radiation dose. The economic benefits of burnup credit result from lower storage, shipping, and disposal costs, and reduced package handling operations at storage, shipping, and receiving facilities.

Burnup credit is already standard practice in borated spent fuel pools. "*Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants*"[ES-1] provides the regulatory position that utilizes burnup credit which includes fission products. The regulatory position depends on boron in the spent fuel pool, whose presence is not credited in the criticality analysis. Although the application of burnup credit in spent fuel pools is different, there are significant similarities that may provide useful guidance.

This topical report describes a methodology for using burnup credit in the design of criticality control systems for PWR spent fuel transportation packages, pursuant to the requirements of 10 CFR Part 71 (also Part 72 for storage/transportation dual-purpose packages). This topical report is expected to be referenced in a number of transportation cask applications to be submitted by commercial cask and canister designers to the NRC. Therefore, NRC acceptance of this topical report will result in increased efficiency of the review process for these SNF cask applications.

The actinide-only burnup credit methodology presented in this report applies to all current generation commercial PWR fuel, with the following restrictions:

- Burnup credit benefits can be gained from fuel irradiated up to 40 GWd/MTU. SNF with an assembly average burnup greater than 40 GWd/MTU shall be treated as having a burnup of 40 GWd/MTU for the purposes of this methodology.

- Enrichments above 4.05 (nominal 4) weight percent U-235 are not considered.

- Assemblies with integral fuel burnable absorbers (IFBAs) are not considered.

- The methodology applies to SNF with cooling times ranging from 1 to 200 years.

- Reconstituted or disassembled fuel is not considered. Also not considered are fuel assemblies which have had any of their original rods removed or replaced.

The burnup credit criticality analysis procedure has been developed to be consistent with the criticality analysis procedure currently accepted by the NRC for which the fresh fuel assumption is made. The purpose of the criticality safety analysis, using the fresh fuel assumption, is to develop a cask loading criterion that establishes the maximum initial enrichment of an SNF assembly design that can be loaded into a cask. The burnup credit criticality analysis procedure builds upon the fresh fuel procedure. The burnup credit procedure results in a SNF package loading criteria that specify minimum burnups necessary for a range of initial enrichment values for a specific fuel assembly design. These results are presented as burnup credit loading curves.

These loading curves make a specific allowance for, and allocation of the various biases and uncertainties in the cask's design burnup capability and the uncertainties in the burnup of the assemblies to be loaded into the cask. Specifically, the loading curve includes various criticality design conservatisms that are part of the methodology, plus a methodology-specified allowance for the uncertainties in the ability of the designer's methods to predict reactivity. To use the burnups on the load curve for selecting and loading individual fuel assemblies, the cask user must verify each candidate assembly's burnup and reduce that verified burnup by the amount of the burnup uncertainty. The topical report does not address the methods of verifying burnup and determining assembly burnup uncertainty. It is anticipated that the user community will continue to develop methods, followed by regulatory review of those methods.

The key elements that distinguish the burnup credit procedure from the fresh fuel procedure and for which NRC acceptance is sought are described below.

Isotopic Validation

The isotopic composition of fresh fuel is well known through extensive, routine measurements by fuel manufacturers. However, after fuel is irradiated in a reactor, the isotopic composition of the spent fuel must be determined through analysis. Routine measurement of the isotopic content of discharged fuel using chemical assays would not be practical. Confidence in the analytical capabilities is high due to the good agreement between the analytical predictions used for core reload analyses and the constant measurements of reactivity and power distributions at power

plants. Source terms generated for thermal and shielding analyses have also agreed well with experiments.

For the burnup credit methodology presented in this topical report, the code system used for predicting isotopic content must be validated. The recommended validation method uses a set of chemical assays for spent fuel. These assays represent benchmarks for which best estimate predictions are computed with the code. The ratio of the measured benchmarks and the computed best estimate predictions are used to determine multiplicative biases and uncertainties. The biases and uncertainties for each isotope are combined in a conservative manner into a correction factor for each isotope. The correction factors are calculated and applied conservatively to ensure that criticality safety evaluations employing the burnup credit method result in a neutron multiplication factor that is conservative for the system being evaluated.

The isotopic validation method is applicable to any computer code system. For the purpose of demonstrating the method, the SAS2H sequence of SCALE 4.2 is used.

This topical report is specifically seeking NRC acceptance of the following:

- The PWR fuel post-irradiation examination assay data selected for isotopic inventory bias and uncertainty determination are sufficient for validating the selected actinide composition in spent fuel

- The statistical procedure proposed for establishing isotope-specific biases and correction factors is a conservative method to account for isotopic concentration changes during burnup.

Criticality Validation

Criticality analysis methods applied in fresh fuel assumption design evaluations are validated by performing benchmark calculations using well-characterized criticality experiments. The burnup credit criticality analysis method is also validated using well-characterized criticality experiments. The criticality validation establishes the validity of the best-estimate calculational method used to determine the effective multiplication factor (k_{eff}) of a system and for deriving the subcritical safety limit consistent with ANSI/ANS-8.1 and ANSI/ANS-8.17 criteria.

The criticality experiment benchmark validation calculations are used to establish method bias and uncertainty over a specific range of package and fuel characteristics. Fresh fuel assumption methods for evaluating PWR applications are typically benchmarked against low enrichment, unirradiated heterogeneous UO_2 fueled systems with similar characteristics to the package being evaluated. The burnup credit method is also benchmarked against UO_2 fueled systems that contain the important U-235 and U-238 burnup credit isotopes. The burnup credit criticality validation also includes low enrichment, unirradiated heterogeneous mixed oxide (MOX) fueled systems. MOX experiments provide benchmark data for other transuranic isotopes present in spent fuel and included in the burnup credit analysis procedure. Burnup credit method bias and uncertainty

results are used to establish the subcritical safety limit to be applied in criticality safety evaluations employing the burnup credit methodology. The subcritical safety limit is calculated based on a statistically determined magnitude of the method biases, uncertainties, and administrative safety margins.

This topical report is specifically seeking NRC acceptance of the following:

- The selection of the 47 MOX critical experiments for actinide-only burnup credit analysis
- The use of the most limiting USL from the UO_2 and MOX criticality experiment analysis
- Confirmation that the methods for criticality validation established in NUREG/CR-5661 applies for burnup credit, with the supplement of the MOX critical experiments as specified above.

Analysis and Modeling Parameters

Analyses performed for validation use best-estimate values to simulate specific experimental conditions. Design basis analyses are more generic and must address a range of parameters. Therefore, all of the key reactor operating parameters in the burnup analysis such as moderator density, soluble boron level, fuel temperature, specific power, and operating history must be conservatively selected at bounding values for actinide-only criticality analysis. These values will serve as limits to the applicability of a given burnup credit design application.

The k_{eff} analysis of the SNF package requires conservatism in the moderator density in the package and the axial profile used for the burnup. The designer is required to perform the package analysis at the most reactive moderator density, and is required to prove that the density selected is the most reactive.

This topical report is specifically seeking NRC acceptance of the following:

- A single cycle burnup at a specific power of 60 MW/MTU conservatively bounds the effects of specific power and operating history on isotopic concentrations
- The use of the maximum cycle average dissolved boron concentration conservatively accounts for soluble boron effects on isotopic concentrations
- The reactivity of the spent fuel is maximized by setting the fuel temperature to the maximum pellet averaged temperature
- The use of the maximum core outlet temperature in determining the moderator density for depletion produces conservative isotopic concentrations
- The method presented for determining optimum moderation in the SNF package is adequate

- The use of the selected limiting axial burnup profiles for burnup credit conservatively capture the end effects

- The selected horizontal gradients and use of the most limiting arrangement in the package analysis sufficiently model horizontal burnup effects.

Spent Nuclear Fuel Package Loading Criteria

The result of performing a burnup credit criticality analysis is the development of burnup credit loading curves. The curves specify the loading criteria, by indicating the minimum burnup necessary for a fuel assembly with a specific initial enrichment and minimum cooling time to be placed in a burnup credit package. Multiple curves may be beneficial or necessary due to variations in fuel assembly designs (i.e., Westinghouse versus B&W designs). In addition, separate curves can be generated for fuel that was irradiated with removable burnable absorbers. The loading curves account for biases and uncertainties in the criticality design of the cask. Biases and uncertainties associated with the declared burnup are separately accounted for in loading procedures developed by the user, and are not addressed in this topical report. This topical report is seeking NRC acceptance of the method used to generate two-parameter loading curves (i.e., burnup and initial enrichment) for specifying package burnup credit loading requirements.

Physical Implementation and Controls

The user of a certified burnup credit transportation cask will be required to implement controls during cask loading to ensure that design basis fuel requirements and licensing conditions related to burnup credit are being satisfied. However, the detailed nature of these controls and their implementation are still being developed. Therefore, the NRC is not being asked to review and accept the controls or their methods of implementation as a part of this topical report. However, there are several reasonable processes for such implementation.

Specifically, it is anticipated that for each PWR assembly to be loaded, the users of NRC-certified burnup credit transportation casks will be required to:

- Verify that each assembly is appropriate to the loading curve being used, specifically with regard to the number of removable burnable poison rods (no greater than the number used for the load curve), and the cooling time (no less than for the load curve, but no longer than 200 years)

- Independently confirm the average burnup of each assembly by a measurement, reduce the verified assembly burnup by the amount of the burnup uncertainty, and verify that the so-reduced assembly average burnup falls on, or above the load curve burnup for that assembly's initial enrichment.

In summary, the NRC is being asked to accept the burnup credit methodology described in this topical report. Following NRC acceptance, the topical report is expected to be used and referenced by commercial cask designers in license applications. The availability of the approved topical report

will thus facilitate both the design and the NRC review processes. Separately, but in parallel with these activities, it is expected that cask vendors, utility service vendors, and utilities will continue working with the NRC to establish acceptable practices for demonstrating that the loaded fuel complies with the burnup credit conditions of cask certification.

1. INTRODUCTION

This chapter provides introductory information on burnup credit and presents an overview of the burnup credit methodology.

1.1 BACKGROUND

The Nuclear Waste Policy Act of 1982 (NWPA), as amended,[1-1] assigns to the United States Department of Energy (DOE) the responsibility for managing the disposal of civilian spent nuclear fuel (SNF) and high-level radioactive waste (HLW). To fulfill this responsibility, the DOE Office of Civilian Radioactive Waste Management (OCRWM) is developing a Civilian Radioactive Waste Management System (CRWMS) to accept, transport, and permanently dispose of the waste. The transport packages that will be used to carry the SNF from commercial utility reactor sites to the CRWMS facilities will be licensed by the United States Nuclear Regulatory Commission (NRC) in accordance with the requirements of Title 10 to the Code of Federal Regulations (CFR) Part 71[1-2] (Packaging and Transportation of Radioactive Material).

To meet 10 CFR Part 71 requirements, SNF transportation packages must be designed to ensure criticality safety. Criticality safety is ensured by package design features, such as maintaining SNF geometry and the use of supplemental neutron absorbing materials, as well as administrative controls. Administrative controls are required to ensure SNF loaded into a transportation package meets design basis fuel requirements and applicable licensing conditions. Design basis fuel requirements and licensing conditions typically include limits on fuel assembly parameters including initial enrichment (for criticality), burnup (for shielding) and cooling time (for shielding).

Historically, safety analyses of criticality control systems for transportation packages include an assumption that the SNF loaded into the package is "fresh" or unirradiated. In other words, the spent fuel is assumed to have its original, as-manufactured U-235 isotopic content. The "fresh fuel" assumption is very conservative since the reactivity of the nuclear fuel is substantially reduced after being irradiated in the reactor core. The concept of taking credit for this reduction in nuclear fuel reactivity due to burnup of the fuel, instead of using the fresh fuel assumption in the criticality safety analysis, is referred to as "burnup credit." Burnup credit uses the actual physical composition of the fuel and accounts for the net reduction of fissile material and the buildup of neutron absorbers in the fuel as it is irradiated. Neutron absorbers include actinides and other isotopes generated as a result of the fission process. Using only the change in actinide isotopes in the burnup credit criticality analysis is referred to as "actinide-only burnup credit."

Although the fresh fuel assumption represents a conservative design approach, which substantially simplifies the criticality safety analysis and associated administrative controls, it results in a significant reduction in SNF capacity for a given package weight. Analyses performed by DOE and its contractors have indicated that using burnup credit to maximize SNF transportation cask capacities is a justifiable concept that would result in public risk benefits and cost savings while fully maintaining criticality safety margins.[1-3] The use of burnup credit in the design of criticality control systems enables more spent fuel to be placed in a package. Increased package capacity

in turn results in reduced environmental impact in the form of a reduced number of containers and related handling and transport operations for a given number of SNF assemblies. Several public and rate payer benefits result from an overall reduction in the number of packages because the total number of packages drives both cost and risk. Fewer shipments result in a lower risk of accidents associated with the handling and transportation of spent fuel, thus reducing both radiological and non-radiological risk to the public. (Although there is a reduction in the radiological risk, this risk is already extremely small.) A reduced number of packages also reduces worker radiation doses and therefore has occupational As Low As is Reasonably Achievable (ALARA) benefits. The economic benefits of burnup credit result from lower storage, shipping, and disposal costs, and reduced package handling operations at storage, shipping, and receiving facilities. Given the large quantity of SNF, and the high costs of the packages, there are substantial incentives for using burnup credit in the design of SNF packages.

Burnup credit is already standard practice in borated spent fuel pools. "*Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants*"[1-4] provides the regulatory position that utilizes burnup credit, which includes fission products. The regulatory position depends on boron in the spent fuel pool, whose presence is not credited in the criticality analysis. Although the application of burnup credit in spent fuel pools is different, there are significant similarities that may provide useful guidance.

This topical report describes a methodology, to be used as guidance, for validating analytical methods and for applying burnup credit in the design of criticality control systems for pressurized water reactor (PWR) spent fuel transportation packages or storage/transport dual-purpose packages. The report references technical data, analyses, and results that have been developed over the years by OCRWM and its contractors in support of burnup credit. The topical report uses and organizes these data and analyses to develop validation and analysis methodologies as well as operational processes necessary for implementation of burnup credit.

1.2 OBJECTIVE

The objective of this topical report is to present to the NRC for review and acceptance a methodology for using burnup credit in the design of criticality control systems for PWR spent fuel transportation packages, while maintaining the criticality safety margins and related requirements of 10 CFR Part 71 and 72. The proposed methodology consists of five major steps as summarized below:

1. Validate a computer code system to calculate isotopic concentrations in SNF created during burnup in the reactor core and subsequent decay.

2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff}, of a spent nuclear fuel package.

3. Establish bounding conditions for the isotopic concentration and criticality calculations.

4. Use the validated codes and bounding conditions to generate package loading criteria (burnup credit loading curves).

5. Verify that SNF assemblies meet the package loading criteria and confirm proper fuel assembly selection prior to loading. (This step is required but the details are outside the scope of this topical report.)

When reviewed and accepted by the NRC, this topical report will serve as a criterion document for criticality control analysts and will provide steps for the use of actinide-only burnup credit in the design of criticality control systems. The NRC-accepted burnup credit methodology will be used by commercial SNF storage and transportation package designers. Design-specific burnup credit criticality analyses will be defined, developed, and documented in the Safety Analysis Report (SAR) for each specific storage or transportation package that uses burnup credit. These SARs will then be submitted to the NRC for review and approval. This topical report is expected to be referenced in a number of storage and transportation cask applications to be submitted by commercial cask and canister designers to the NRC. Therefore, NRC acceptance of this topical report will result in increased efficiency of the review process for these SNF storage and transportation cask applications. The DOE will also reference NRC-accepted topical reports in its license application for a geologic repository.

DOE is requesting NRC acceptance for two general aspects of the actinide-only burnup credit methodology. First, data is sufficient to validate the burnup credit criticality analysis methodology presented in this topical report. This includes the chemical assay data used to validate the spent fuel isotopic concentration calculations and critical experiments used to validate the burnup credit criticality calculations. Second, the conservative methodology in utilizing this data for burnup credit is acceptable. A detailed breakdown of what the DOE is specifically seeking NRC acceptance of is presented in Section 1.6.

1.3 SCOPE

This topical report presents a methodology for using actinide-only burnup credit in the design of PWR spent fuel packages. For completeness, it also summarizes the type of requirements that need to be met by cask users when loading SNF into a transportation package that has been licensed for burnup credit. NRC approval is not being requested for these latter requirements. Actinide-only burnup credit addresses just the reduced reactivity of SNF due to changes in actinide isotopes. The considerable additional negative reactivity effect of fission products is used as a margin for gaining consensus on the methodology.

The actinide-only burnup credit methodology presented in this report has a wide applicability. It applies to all current generation commercial PWR fuel, with the following restrictions:

- Burnup credit benefits can be gained from fuel burned up to 40 GWd/MTU. SNF with an assembly average burnup greater than 40 GWd/MTU shall be treated as having a burnup of 40 GWd/MTU for the purposes of this methodology.

- Enrichments above 4.05 (nominal 4) weight percent U-235 are not considered.
- Assemblies with integral fuel burnable absorbers (IFBAs) are not considered.
- The methodology applies to SNF with cooling times ranging from 1 to 200 years.
- Mixed oxide (MOX) initial content fuel is not considered.
- Reconstituted or disassembled fuel is not considered. Also not considered are fuel assemblies which have had any of their original rods removed or replaced.

The isotopic validation also includes limits on the range of applicability based on spectral index and specific power parameters. Both the spectral index and specific power have a range that covers all commercial PWR SNF.

There are analysis and modeling parameters that affect criticality, which are not unique to burnup credit. None of these parameters or effects impact the proposed burnup credit methodology; therefore, they are not included in this topical. A licensee's Safety Analysis Report is required to address these parameters in the usual manner. Examples include:

- Material and fabrication tolerances
- Uncertainties due to limitations in the geometric or material representations used in the computational method
- Effects of symmetric or asymmetric fuel assembly clustering within the spent fuel basket.

1.4 REGULATORY REQUIREMENTS

Compliance with NRC regulatory requirements is accomplished by applying available regulatory guidance, industry standards, and regulatory precedent established by previous certification applications. Criticality safety design criteria are set forth in the Code of Federal Regulations. In addition to the NRC regulations, NRC Regulatory Guides (RGs) address criticality safety. These RGs have been considered for applicability to the burnup credit methodology discussed in this report. The RGs typically accept the procedures and methodologies developed in ANSI/ANS Standards. ANSI/ANS Standards provide basic recommendations that can be referenced or used with other safety standards or regulations to address criticality safety requirements. The sections below discuss the specific NRC regulatory requirements and industry guidance upon which the burnup credit topical report is based.

1.4.1 Criticality Safety Design Criteria

The NRC regulatory requirements for transportation and storage of SNF are established in 10 CFR Parts 71 and 72. A design criterion which is key to this regulation is nuclear criticality safety. Nuclear criticality safety criteria for the design and certification of SNF transportation packages are set forth in 10 CFR § 71.55(b), (d), and (e), § 71.57 and 10 CFR § 72.124.

The burnup credit methodology presented in this topical report is consistent with the general design criteria specified in 10 CFR Parts 71 and 72. Section 1.3 discusses the scope and specific restrictions imposed on the proposed burnup credit methodology.

1.4.2 Applicable Regulatory Guides and Standards

Outlined below are the Regulatory Guides and ANSI/ANS Standards whose guidance has been incorporated into the burnup credit methodology.

- Regulatory Guide 3.4, *Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Material Facilities.*[1-5] This Regulatory Guide endorses ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.*[1-6] The burnup credit topical report complies with guidance provided in RG 3.4 and ANSI/ANS-8.1 as discussed in Chapter 3 of this report.

- Regulatory Guide 3.58, *Criticality Safety for Handling, Storing, and Transporting LWR Fuel at Fuels and Materials Facilities.*[1-7] This Regulatory Guide endorses ANSI/ANS-8.17, *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*[1-8] with the following exception. The Regulatory Guide states that credit for fuel burnup may be taken only when the amount of burnup is confirmed by reactivity measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored. The burnup credit topical report complies with the guidance provided in RG 3.58, but not with regard to this exception. Instead, the burnup credit topical report complies with the guidance of ANSI/ANS-8.17, which allows credit for fuel burnup by analysis, as discussed in Chapters 2, 3, 4, and 5, and by a measurement-based verification of the exposure history of each fuel assembly, as discussed in Chapter 6 of this report.

- Regulatory Guide 3.60, *Design of an Independent Spent Fuel Storage Installation (Dry Storage).*[1-9] The Regulatory Guide endorses ANSI/ANS-57.9, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).*[1-10] Consideration has been given to the guidance in Regulatory Guide 3.60 and ANSI/ANS 57.9 in the development of the burnup credit topical report. With regard to Criticality Safety, 57.9 endorses ANSI/ANS-8.17, which allows credit for fuel burnup by analysis and verification of the exposure history. Therefore, the burnup credit topical report complies with the applicable guidance provided in RG 3.60 and ANSI/ANS-57.9

- Draft Regulatory Guide 1.13, *Proposed Revision 2 to Regulatory Guide 1.13 Spent Fuel Storage Facility Design Basis*.[1-11] This draft Regulatory Guide endorses ANSI/ANS-57.2, *Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants*,[1-12] subject to several clarifications and modifications. The burnup credit topical report complies with the applicable guidance contained in draft RG 1.13 and ANSI/ANS-57.2 as discussed in Chapter 6 of this report.

1.5 QUALITY ASSURANCE

The Quality Assurance Requirements and Description (QARD) document is the principal quality assurance document for the Civilian Radioactive Waste Management program.[1-13] The QARD meets the applicable QA program requirements in 10 CFR Part 50, Appendix B; 10 CFR Part 71; 10 CFR Part 72; 10 CFR Part 60; and NQA-1. The QARD establishes the QA requirements for OCRWM, contractors, subcontractors, national laboratories, and other Government agencies performing activities for OCRWM that are quality affecting.

The key elements of the QARD are standard. They require planning, identification of inputs, identification of assumptions, thorough analysis by qualified analysts, checking, and documentation. Analyses performed are sufficiently detailed as to the purpose, method, assumptions, input, and references such that a person technically qualified in the subject can understand the analysis and verify its adequacy without recourse to the originator. Technical analysis outputs specify the appropriate level of inspection and testing necessary to ensure technical adequacy. Technical document reviews are performed to ensure that the inputs are correctly selected for their incorporation into the analysis. Assumptions are described and where applicable, identified as requiring additional confirmation as the design proceeds. The technical outputs are reasonable compared to the inputs, and necessary technical input for interfacing organizations are specified in the documents. QA records are legible, accurate, and completed appropriate to the work accomplished. Records are indexed for ease in retrieval. Records are distributed, handled, and controlled in accordance with the QA procedures. This includes proper identification, classification, distribution, storage, retrieval, and disposition. The process is subject to QA audits to ensure compliance with the applicable procedures.

Much of the analysis in support of this document was performed by the Management and Operating Contractor (M&O) following the procedures written to support Section 3.0, Design Control, of the QARD. No tests or experiments were performed by the M&O.

This topical report references technical data, analyses, and results that have been developed by OCRWM contractors. Where applicable, these reference documents have been developed under the respective contractor QA programs in compliance with OCRWM's QA program. Some data used in the development of the burnup credit criticality analysis procedure are derived from reports, experiments, or records that are not subject to the requirements of OCRWM's QA program. The qualification of these data is addressed in the appropriate sections of the topical report.

1.6 OVERVIEW OF THE BURNUP CREDIT CRITICALITY ANALYSIS METHODOLOGY

The burnup credit criticality analysis procedure has been developed to be consistent with the criticality analysis procedure, currently accepted by the NRC, for which the fresh fuel assumption is made. The generic criticality safety analysis procedure using the fresh fuel assumption is illustrated in Figure 1-1. The purpose of the criticality safety analysis using the fresh fuel assumption is to develop a cask loading criterion that establishes the maximum initial enrichment of an SNF assembly design that can be loaded into a cask. Figure 1-2 illustrates the generic burnup credit criticality analysis procedure recommended in this topical report. The burnup credit criticality analysis procedure builds upon the fresh fuel procedure. The burnup credit procedure results in spent nuclear fuel package loading criteria that specify minimum burnups necessary for a range of initial enrichment values for a specific fuel assembly design. These results are presented as burnup credit loading curves.

The key elements in Figure 1-2 that distinguish the burnup credit procedure from the fresh fuel procedure are shaded. NRC acceptance is sought for these key elements of the burnup credit procedure, which are briefly described in the following subsections. Detailed descriptions of each of these elements and their relevance to the regulatory requirements are provided in the body of the topical report. In the following discussion, refer to Figure 1-2 for an understanding of where these key elements fit into the overall burnup credit criticality analysis procedure.

1.6.1 Isotopic Validation

The isotopic composition of fresh fuel is well known through extensive, routine measurements by fuel manufacturers. However, after fuel is irradiated in a reactor, the isotopic composition of the spent fuel is routinely determined through analysis, rather than through measurement. Routine measurement of the isotopic content of discharged fuel using chemical assays is not practical due to dose, safety, and cost concerns. Confidence in the analytical capabilities is high due to the good agreement between the analytical predictions used for core reload analyses and the continual measurements of reactivity and power distributions at power plants. Source terms generated for thermal analyses have also shown good agreement with experiments.

For the burnup credit methodology presented in this topical report, the computer code system used for predicting isotopic content must be validated. The recommended validation method uses a set of chemical assays for spent fuel. These assays represent measured data for which best estimate predictions are analyzed with the computer code. The ratio of the measured benchmarks and the computed best estimate predictions are used to determine multiplicative biases and uncertainties. The biases and uncertainties for each isotope are combined in a conservative manner into a correction factor for each isotope. The correction factors are calculated and applied conservatively to ensure that criticality safety evaluations employing the burnup credit method result in a neutron multiplication factor that is conservative for the system being evaluated.

The isotopic validation method is applicable to any computer code system. For the purpose of demonstrating the method, the SAS2H sequence of SCALE 4.2[1-14] is used.

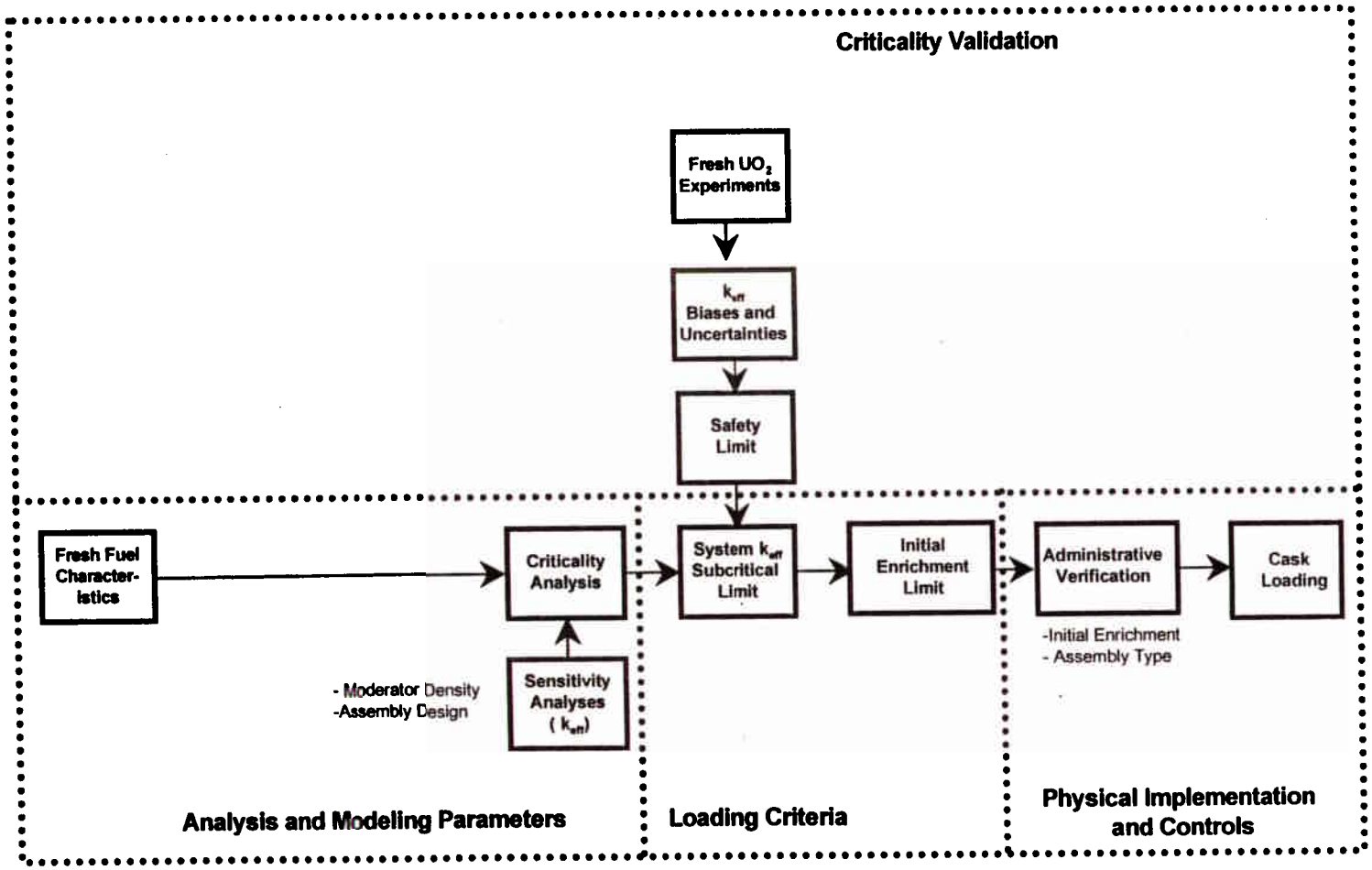

Figure 1-1. Fresh Fuel Assumption Procedure

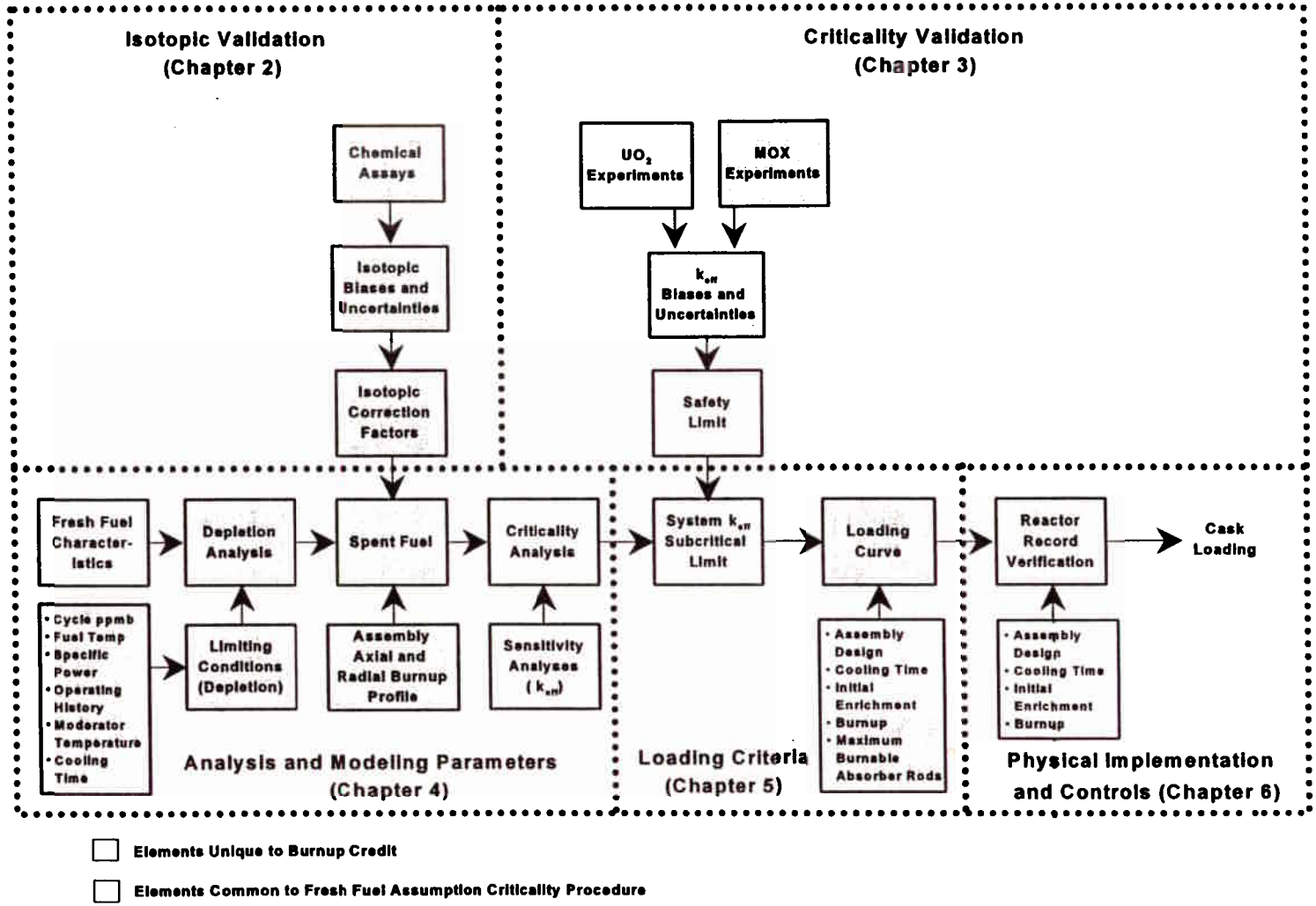

Figure 1-2. Burnup Credit Procedure

This topical report is specifically seeking NRC acceptance of the following:

- The PWR fuel post-irradiation examination assay data selected for isotopic inventory bias and uncertainty determination is sufficient for validating the selected actinide composition in spent fuel

- The statistical procedure proposed for establishing isotope-specific biases and correction factors results in a conservative method to account for isotopic concentration changes during burnup.

1.6.2 Criticality Validation

Criticality analysis methods applied in fresh fuel assumption design evaluations are validated by performing benchmark calculations using well-characterized criticality experiments. The burnup credit criticality analysis method is also validated using well-characterized criticality experiments. The criticality validation establishes the validity of the best-estimate calculational method used to determine the effective multiplication factor (k_{eff}) of a system and for deriving the subcritical safety limit consistent with ANSI/ANS-8.1 and ANSI/ANS-8.17 criteria.

The criticality experiment benchmark validation calculations are used to establish method bias and uncertainty over a specific range of package and fuel characteristics. Fresh fuel assumption methods for evaluating PWR applications are typically benchmarked against low enrichment, unirradiated heterogeneous UO_2 fueled systems with similar characteristics to the package being evaluated. The burnup credit method is additionally benchmarked with low enrichment, unirradiated heterogeneous mixed oxide (MOX) fueled systems. MOX experiments provide benchmark data for other transuranic isotopes present in spent fuel and included in the burnup credit analysis procedure. Burnup credit method bias and uncertainty results are used to establish the subcritical safety limit to be applied in criticality safety evaluations employing the burnup credit methodology. The subcritical safety limit is calculated based on a statistically determined magnitude of the method biases, uncertainties, and administrative safety margins.

This topical report is specifically seeking NRC acceptance of the following:

- The selection of the 47 MOX critical experiments for actinide-only burnup credit analysis

- The use of the most limiting Upper Subcriticality Limit (USL) from the UO_2 and MOX criticality experiment analysis

- Confirmation that the methods for criticality validation established in NUREG/CR-5661 apply to burnup credit, with the supplement of the MOX critical experiments as specified above.

1.6.3 Analysis and Modeling Parameters

Analyses performed for validation use best-estimate values to simulate specific experimental conditions. Design basis analyses are more generic and must address a range of parameters. Therefore, all of the key reactor operating parameters in the burnup analysis, such as moderator density, soluble boron level, fuel temperature, specific power, and operating history, have been conservatively selected at bounding values for actinide-only criticality analysis. These values will serve as limits to the applicability of a given burnup credit design application.

The k_{eff} analysis of the SNF package is based on conservatism in the moderator density in the package and the axial profile used for the burnup. To maintain this conservatism, the designer is required to perform the package analysis at the most reactive moderator density and is required to demonstrate that the density selected is the most reactive.

This topical report is specifically seeking NRC acceptance of the following:

- A single cycle burnup at a specific power of 60 MW/MTU conservatively bounds the effects of specific power and operating history on isotopic concentrations
- The use of the maximum cycle average dissolved boron concentration conservatively accounts for soluble boron effects on isotopic concentrations
- The reactivity of the spent fuel is maximized by setting the fuel temperature to the maximum pellet averaged temperature
- The use of the maximum core outlet temperature in determining the moderator density for depletion produces conservative isotopic concentrations
- The method presented for identifying and demonstrating optimum moderation in the SNF package is adequate
- The use of the selected limiting axial burnup profiles for burnup credit conservatively captures the end effects
- The selected horizontal gradients and use of the most limiting arrangement in the package analysis sufficiently model horizontal burnup effects.

1.6.4 Spent Nuclear Fuel Package Loading Criteria

The result of performing a burnup credit criticality analysis is the development of burnup credit loading curves. The curves specify the loading criteria by indicating the minimum burnup necessary for a fuel assembly with a specific initial enrichment and minimum cooling time to be placed in a burnup credit package. Multiple curves may be necessary due to variations in fuel

assembly designs (i.e., Westinghouse versus B&W designs). Also, separate curves may be generated for fuel that was irradiated with removable burnable absorbers.

The development of a burnup credit loading curve is accomplished by performing a set of criticality analyses for a range of initial enrichment. First, the criticality analysis is performed to determine the k_{eff} value for a given initial enrichment and an initial estimate of the required burnup. Then, the burnup is adjusted and the criticality analysis is repeated until a k_{eff} value equal to or less than the allowable value is obtained. The minimum burnup, which results in an acceptable k_{eff} value for the given initial enrichment, is the required minimum burnup. The procedure is repeated for a range of initial enrichments.

This topical report is seeking NRC acceptance of the above method used to generate two-parameter loading curves (i.e., burnup and initial enrichment) for specifying package burnup credit loading requirements.

In dealing with the various reactivity biases and uncertainties that must be addressed in burnup credit it is necessary to allocate the treatment of these uncertainties between two activities. The first activity is the development of the cask loading curve provided by the cask vendor. This is performed by the cask vendor by use of the methodology presented in this topical report. The second activity is the cask loading, conducted by the cask user. This is performed by the cask user through use of a burnup verification measurement and cask user's assembly burnup data. The allocation that has been made in this topical is to include within the loading curve, methodology-specified allowances for the biases and uncertainties in the ability of the designer's methods to predict reactivity. This is the part of the methodology for which NRC approval is being sought. The uncertainties arising from the burnup verification process and the allowance for burnup uncertainty need to be addressed as part of the cask loading process. Although discussed for perspective, no approvals are sought for the loading process.

1.6.5 Physical Implementation and Controls

The loading of SNF transportation packages designed for burnup credit requires the implementation of additional controls during loading to ensure design basis fuel requirements and licensing conditions are met. These controls are in addition to those that are already being implemented for fresh-fuel based packages. ANSI/ANS-8.17 indicates that credit may be taken for fuel burnup by establishing a maximum spent fuel reactivity and ensuring that each fuel assembly has a reactivity no greater than the maximum established by "analysis and verification of the exposure history of each fuel unit." For completeness, this topical report presents a discussion of loading procedures, including the use of measurements to confirm assembly burnup and the use of an allowance for assembly burnup uncertainty, but is not requesting any approvals for the loading or burnup verification process.

1.7 ORGANIZATION OF THE REPORT

The contents of this report are organized following the sequence described in the preceding section and illustrated in Figure 1-2. Chapter 1 provides introductory information on burnup credit and gives an overview of the burnup credit methodology.

Chapter 2 addresses isotopic validation in detail. The available chemical assays of spent fuel isotopics are presented. The actual statistical approach to yield biases and correction factors is then discussed. The SAS2H sequence of the SCALE 4.2 code package with 27BURNUPLIB is used to demonstrate the validation procedure. This chapter also presents the biases and correction factors that result from using the code package.

Chapter 3 endorses the NUREG entitled, "*Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages*," NUREG/CR-5661, and follows its presentation of the criticality validation by noting the changes required for actinide-only burnup credit. The chapter presents MOX criticality benchmark experiments to be used to validate burnup credit criticality calculations. It then develops the method to convert analyses of critical experiments to an upper safety limit on k_{eff}.

Chapter 4 develops guidelines to determine default or bounding values of physical parameters to be used in the analysis of spent fuel compositions and reactivity. The appropriate treatment of the axial burnup is developed in this chapter.

The analytical methods and parameters presented in the previous chapters are combined in Chapter 5 where the generation of burnup credit loading curves is presented. This chapter describes how the loading curves are actually generated.

Chapter 6 discusses the steps and procedures for selecting and loading SNF into a burnup credit package; however, NRC approval of these is not being requested.

Chapter 7 summarizes the topical report and reviews the steps for implementing burnup credit. It also reviews the conservatism in the methodology and the use of some of that margin for gaining consensus with the methodology.

Chapter 8 provides bibliographic information for references.

Appendix A contains an acronym list, and Appendix B is a glossary of terms.

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2. ISOTOPIC VALIDATION

As illustrated in Figure 1-2, it is SNF depletion analysis that differentiates the burnup credit approach from the traditional fresh fuel approach in designing the criticality control systems of spent fuel packages. The depletion analysis simulates the burnup of the fuel under reactor operating conditions. The result of the depletion analysis is the predicted isotopic composition of the discharged spent fuel assembly.

The primary focus of this chapter is on developing a methodology for validating isotopic depletion/generation computer codes used in predicting the quantities of actinide isotopes in SNF. The validation is performed by comparing the calculated to the measured isotopic values. The bias and uncertainty determined from this comparison are then established for each isotope. Subsequently, the bias and uncertainty terms are used to calculate a set of conservative correction factors to be used to modify the isotopic inventory for criticality analysis.

In summary, this chapter describes the process for determining a conservative estimate of concentrations of selected actinide isotopes for use in criticality safety analyses. The major discussions in this chapter are: 1) selecting isotopes to represent spent fuel for criticality analyses; 2) the measured chemical assay data used in the validation of the calculational method; and 3) the method used to establish calculational bias, uncertainty, and correction factors.

2.1 ISOTOPIC SELECTION FOR SPENT FUEL REPRESENTATION

Approximately 1,300 different isotopes are generated in the spent fuel. Representing all these isotopes in an analytical model for criticality analysis is neither practical nor essential. Therefore, a limited set of radionuclides is proposed for the analysis of SNF reactivity.

In making a conservative selection of isotopes to represent the spent fuel composition, the neutron absorption properties of the isotopes should be considered. The concentration and hence the contribution of these isotopes to neutron absorption, resulting in either fission or simple neutron-capture reactions, is dependent on cooling time. Figures 2-1 through 2-3 provide the results of a sensitivity study[2-1] showing the fractional absorption rate as a function of time for the key actinide isotopes.

The isotopes that have a significant positive reactivity worth (U-235, Pu-239, and Pu-241) must be included in the burnup credit methodology. Factors to be considered in conservatively eliminating isotopes with negative worth are the chemical form, physical form and characteristics, solubility, volatility, and verifiability of the isotope by comparison to physical measurements. These factors do not disqualify any of the selected actinides. Although not disqualified by these criteria, consideration of Np-237 and U-236 has been deferred. Np-237 is not considered at this time due to large deviations between the calculated and measured values, and U-236 is not

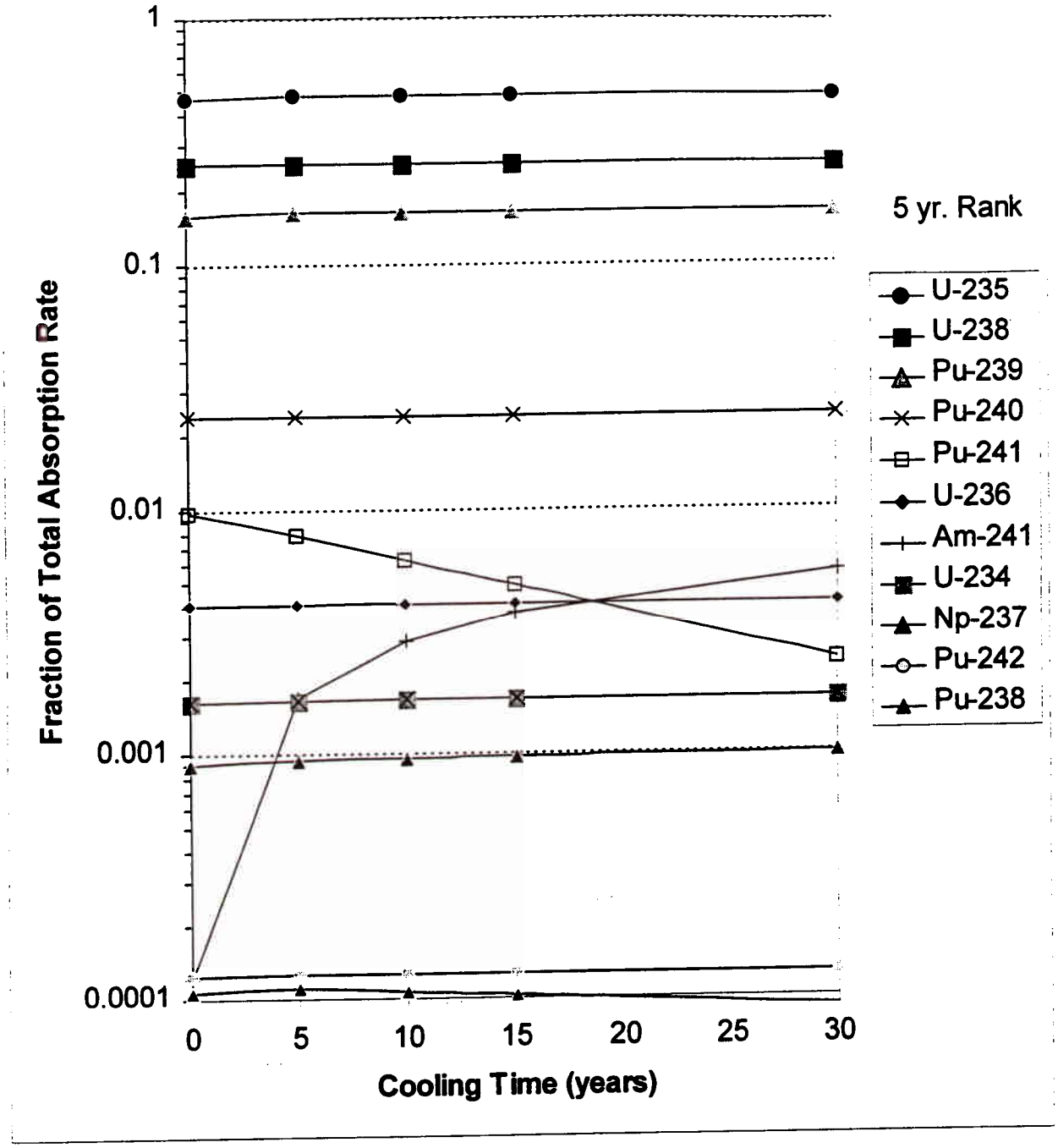


Figure 2-1. Fractions of Neutrons Absorbed by Major Actinides at Various Cooling Times, 3.6 wt.% U-235, 10 GWd/MTU

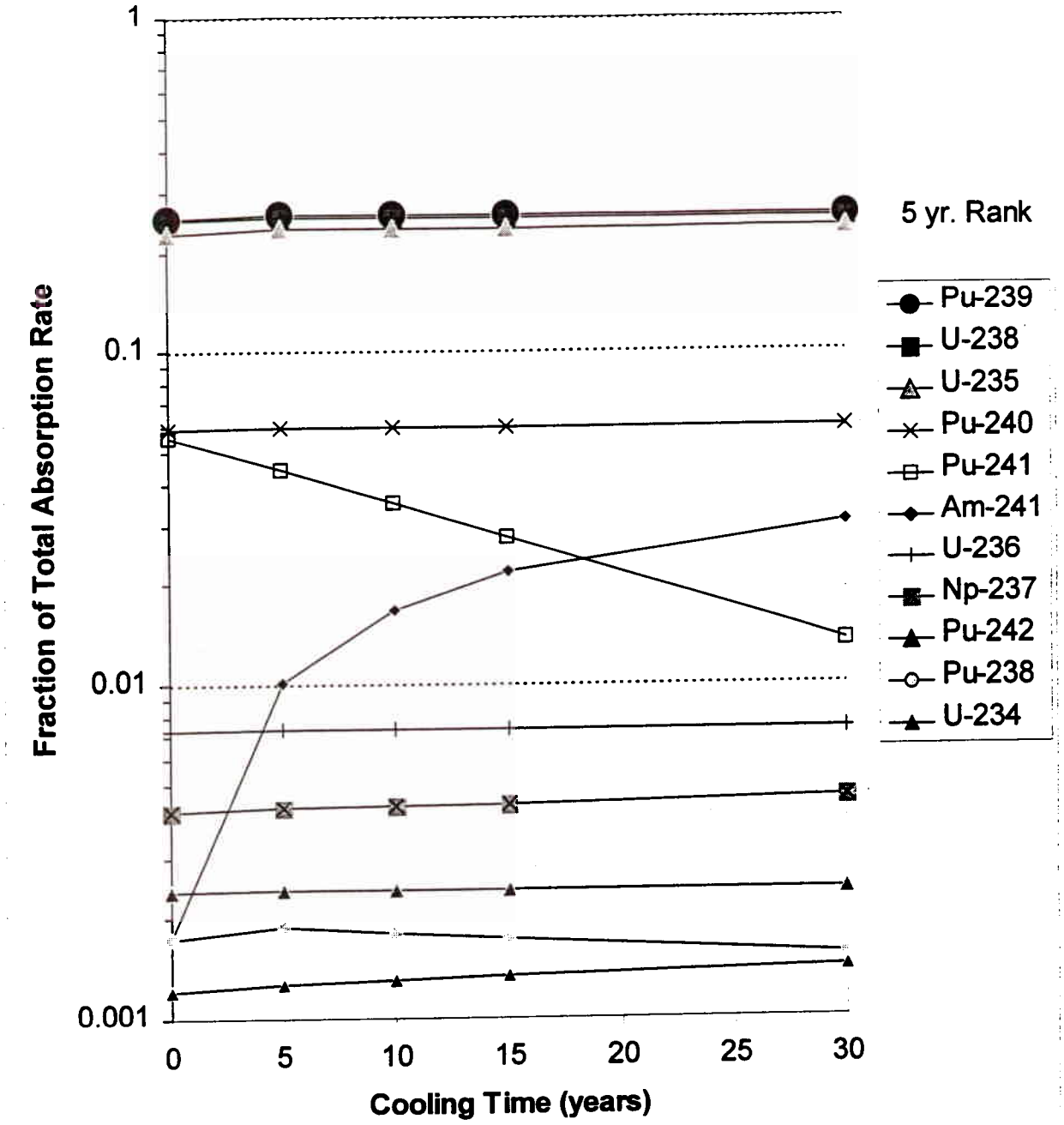


Figure 2-2. Fractions of Neutrons Absorbed by Major Actinides at Various Cooling Times, 3.6 wt.% U-235, 30 GWd/MTU

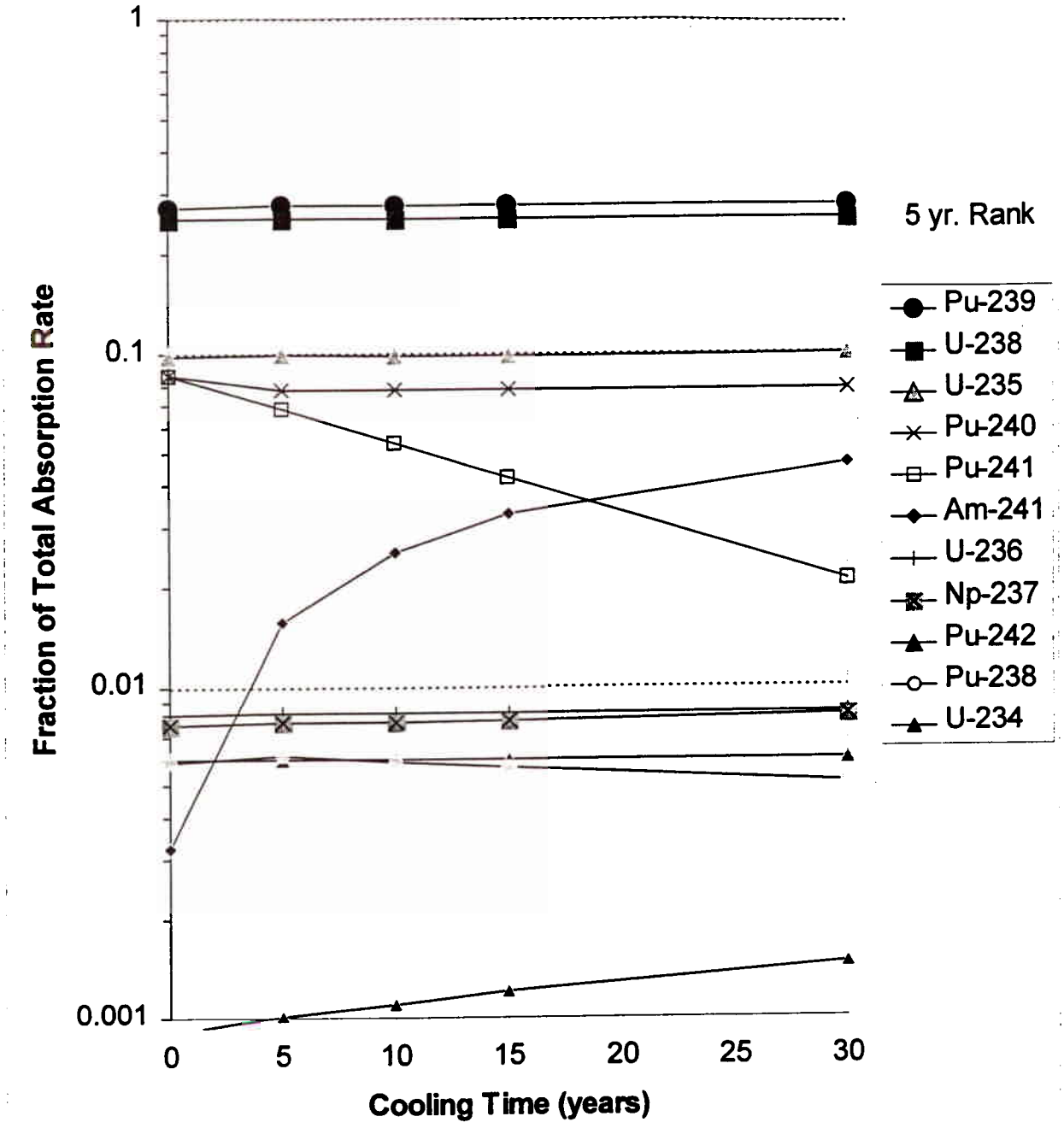


Figure 2-3. Fractions of Neutrons Absorbed by Major Actinides at Various Cooling Times, 3.6 wt.% U-235, 50 GWd/MTU

included due to lack of reactivity worth validation evidence. Therefore, the selected actinide isotopes to be included in the burnup credit criticality analysis methodology reported here are U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241.

2.2 ISOTOPIC VALIDATION DATA

To ensure the accuracy of any computational tool used to predict the isotopic composition in spent fuel, the tool must be validated against a set of measurements performed on spent fuel samples. These samples should test the capabilities of the computational tool over a wide range of parameters important to the isotopic changes in the fuel assemblies. The following subsections present: 1) the isotopic measurement data to be used to validate isotopic computational tools for predicting the selected actinide inventories in spent fuel, 2) an examination of the range of applicability of the data, and 3) the qualification of the data.

2.2.1 Isotopic Validation Measurements

This subsection presents the experimental data recommended for use in validating the calculation of selected actinides in spent fuel. The sources of these data are the Materials Characterization Center (MCC) at the Pacific Northwest Laboratories (PNL),[2-2, 2-3, 2-4, 2-5] data from the German Obrigheim reactor fuel assemblies,[2-6, 2-7] data from Mihama-3, Yankee Rowe, Trino Vercellese, and Turkey Point fuel assemblies.[2-8] A compilation of all the measurements, along with details of benchmark calculations, is provided in References 2-9, 2-10, and 2-11. The following paragraphs provide a summary discussion of the chemical assay data.

The fuel assemblies analyzed at the MCC, which were part of a program to collect information on spent fuel for the Yucca Mountain Repository Project, consisted of three 14x14 Combustion Engineering (CE) assemblies from the Calvert Cliffs Unit 1 reactor and one 15x15 Westinghouse assembly from the H. B. Robinson Unit 2 reactor. From each assembly, a specific fuel pin was selected for the study. The data from the MCC on these assemblies included detailed fuel information collected before the assemblies were destructively assayed. These data included reactor, assembly, and fuel pin specifications; irradiation histories; a description of unusual events that occurred during each assembly's lifetime; burnup measurements; and detailed axial scans using gamma spectroscopy. The radiochemical assays were performed on individual fuel pellets taken from multiple axial positions in each fuel rod to evaluate a distribution of burnups. For each pellet, measurements were performed for the major actinides, cesium isotopes, and Tc-99. The uncertainty for each type of analytical measurement was included in the data documentation. Although a few of the isotopic measurements had large uncertainties, the measurement uncertainties for the selected actinides were approximately 1.6%. The description of Calvert Cliffs and H. B. Robinson spent fuel samples and their calculational models are provided in Reference 2-9.

The Obrigheim data were chosen because they represented assembly-averaged data. Five fuel assemblies were physically divided into full-length halves. Each 12-foot half-section was then dissolved and assayed. Since the MCC provides pellet-specific data, it was decided to add diversity to the benchmark set by adding the Obrigheim assembly-averaged data to the benchmark set. The results from the dissolved assembly analyses provide "assembly-averaged" isotopic values. Obrigheim data are based on samples that were independently evaluated at four different European laboratories. The complete description of Obrigheim spent fuel samples and calculational models is also provided in Reference 2-9.

In the mid 1960's, extensive post-irradiation examinations, including radiochemical analyses, were performed on a selected set of spent fuel assemblies with relatively high enrichment (i.e., 3.4 wt.% U-235), discharged from Yankee Rowe Cores I, II, and IV. The overall purpose of the program was to further the knowledge of physical processes that occur within an operating reactor, and thereby contribute to the advancement of competitive atomic power. The program was carried out under three phases. Under Phase I, 56 fuel rods were removed from 14 Core I fuel assemblies, and in Phase II seven fuel rods were removed from a two-cycled burned assembly. The burnup for the rods from Core I and II ranged from 13 to 18 and 10 to 31 GWd/MTU, respectively. Under Phase III, eight rods from one assembly which had been cycled in Cores I, II, and IV were selected for post-irradiation examinations. The maximum burnup of these rods was 46 GWd/MTU. Because of the relatively high burnup and enrichment, it was decided to use the Yankee Rowe measurement data from a selected number of rods from the assembly which had been cycled in Cores I, II, and IV for this isotopic validation. In addition, Yankee Rowe Core design is significantly different from the typical PWR. Therefore, data from fuel assemblies exposed to a non-typical spectra are included in the benchmark set. The complete description of spent fuel samples and calculational models is provided in Reference 2-9.

As part of an effort by Japan "to obtain quantitative data concerning the characteristics of the high burnup spent fuel dissolution for reprocessing,"[2-8] spent fuel samples from Mihama-3 fuel assembly had been analyzed and reported in Reference 2-8. These samples provide data points at 3.2 wt.% enrichment. Eight Mihama-3 samples with a burnup range of 6.9 - 34.2 GWd/MTU were used as part of a benchmark set for this isotopic validation. The complete description of spent fuel samples and calculational models is provided in Reference 2-9.

Fourteen samples obtained from three spent fuel assemblies discharged from Trino Vercellese (Italy) PWR also were used as part of the benchmark set. The purpose of the Trino Vercellese program was to provide data for isotopic benchmarking purposes. Data from Trino Vercellese was selected for this study because of the relatively high (3.9 wt.%) enrichment for one fuel sample and because of its significantly different core design. The isotopic measurements for Trino Vercellese spent fuel were conducted by the Ispara (Italy) and Karlsruhe (Germany) facilities of the European Joint Research Center. The description of spent fuel samples and calculational models is provided in Reference 2-9.

As part of the pre-test characterization of the assemblies for the Climax - Spent Fuel Test which involved storage of spent fuel assemblies in a granite formation, five fuel rods from the Turkey Point Unit 3 reactor were destructively examined. The results of isotopic measurements made on the five samples taken from the five fuel rods are used as part of this isotopic validation benchmark set. Turkey Point was selected simply because of detailed data that were readily available. Burnup analyses for the five samples were performed by Battelle Columbus Laboratory (BCL) with direction provided by the Hanford Engineering Development Laboratory (HEDL). The description of the fuel samples and calculational models is provided in Reference 2-9.

A total of 54 samples representing 18 different fuel assemblies discharged from 7 different reactors have been used as the benchmark set for actinide-only isotopic validation. Table 2-1 presents the results of the measurements and calculated values for nine actinides.

Since almost all the Am-241 in SNF with at least 5 years cooling time comes from post irradiation decay of Pu-241, the Am-241 produced during depletion will be ignored. The Am-241 isotopic concentration will be based only on the decay of Pu-241 after the incore depletion. The predicted Am-241 values will be biased based on Pu-241 measurements. Therefore, no Am-241 measurements were used to determine the bias associated with predicting Am-241.

In some cases, specific isotopic measurements were not performed or were not reported on all the eight isotopes in a given sample. Thus, the measured data range from 25 to 54 samples per actinide isotope. Reference 2-9 summarizes the complete set of the isotopic measurements for the selected actinides.

2.2.2 Range of Applicability of Measured Data

As prescribed in ANSI/ANS-8.1,[2-12] the validation of a computational method requires that the area of applicability of the data be defined. This section identifies the parameters that influence the isotopic concentrations and then uses those parameters to establish the range of applicability associated with the selected chemical assay data presented in the previous section.

The range of applicability for criticality is normally done by comparing the physical description of the critical assemblies with the package's physical description. This is necessary since each component in the physical description can cause a neutron loss which directly impacts k_{eff}. For isotopic concentration validation, only the concentration of the isotope of interest, its precursor's concentrations, the absorption and capture cross sections of both the isotope of interest and the precursors, flux, and time have any impact on the final concentrations. Therefore, physical descriptions are of much less interest in validating isotopic concentrations than they are in criticality validation since only the fuel material has a direct impact on the isotopic concentrations. Although there is not the direct interest in describing the range of applicability through comparison of physical properties, it is desirable to have strong similarities. Hence, Table 2-2

Table 2-1. Calculated and Measured Isotopic Concentrations for the Selected Actinides (gram/MTU)

		Yankee Rowe								Mihama							
Isotope	Axial Location (cm)	220.2	138.9	57.7	17.0	138.9	57.7	138.9	57.7	18.3	316.4	142.3	236.4	16.4	106.9	32.6	239.9
	Burnup (GWd/MTU)	15.95	30.39	31.33	20.19	32.03	31.41	35.97	35.26	8.30	6.92	15.36	21.29	29.50	32.20	33.71	34.32
	Enrichment (wt-% U-235)	3.40	3.40	3.40	3.40	3.40	3.40	3.40	3.40	3.21	3.21	3.20	3.20	3.21	3.21	3.21	3.21
	ALA	15.62	15.58	15.62	15.70	15.57	15.62	15.56	15.61	16.41	16.19	16.31	16.17	16.31	16.26	16.30	16.12
	Specific Power (MW/MTU)	14.22	27.09	27.93	18.00	28.55	28.00	32.07	31.43	38.60	32.19	28.34	39.28	32.00	34.92	36.56	37.22
U-234	C	1.57e+02	1.27e+02	1.24e+02	1.47e+02	1.22e+02	1.24e+02	1.15e+02	1.16e+02	2.55e+02					1.86e+02	1.82e+02	
	M	1.55e+02	1.42e+02	1.19e+02	1.44e+02	1.14e+02	1.15e+02	1.18e+02	1.20e+02	2.67e+02					1.74e+02	1.74e+02	
U-235	C	1.96e+04	1.18e+04	1.13e+04	1.68e+04	1.11e+04	1.13e+04	9.61e+03	9.77e+03	2.34e+04	2.47e+04	1.76e+04	1.38e+04	9.33e+03	8.24e+03	7.58e+03	7.67e+03
	M	1.98e+04	1.26e+04	1.19e+04	1.72e+04	1.18e+04	1.19e+04	9.78e+03	9.84e+03	2.39e+04	2.52e+04	1.85e+04	1.45e+04	9.79e+03	9.07e+03	7.89e+03	8.04e+03
U-236	C	2.86e+03	4.11e+03	4.17e+03	3.32e+03	4.21e+03	4.17e+03	4.40e+03	4.37e+03	1.75e+03	1.54e+03	2.64e+03	3.26e+03	4.10e+03	4.24e+03	4.31e+03	4.33e+03
	M	2.88e+03	4.08e+03	4.15e+03	3.30e+03	4.18e+03	4.09e+03	4.45e+03	4.44e+03	1.65e+03	1.46e+03	2.65e+03	3.27e+03	3.83e+03	4.00e+03	4.18e+03	4.20e+03
U-238	C	9.53e+05	9.40e+05	9.40e+05	9.50e+05	9.39e+05	9.40e+05	9.35e+05	9.36e+05	9.62e+05	9.63e+05	9.57e+05	9.52e+05	9.46e+05	9.44e+05	9.43e+05	9.41e+05
	M	9.49e+05	9.37e+05	9.35e+05	9.36e+05	9.35e+05	9.36e+05	9.33e+05	9.34e+05	9.62e+05	9.62e+05	9.56e+05	9.52e+05	9.47e+05	9.44e+05	9.44e+05	9.42e+05
Pu-238	C	4.07e+01	1.83e+02	1.90e+02	6.95e+01	2.01e+02	1.91e+02	2.57e+02	2.44e+02	4.93e+00	3.58e+00	2.37e+01	5.41e+01	1.24e+02	1.52e+02	1.66e+02	1.82e+02
	M	4.73e+01	1.76e+02	2.14e+02	7.91e+01	2.22e+02	2.11e+02	2.47e+02	2.40e+02	4.51e+00	3.41e+00	2.59e+01	5.72e+01	1.31e+02	1.59e+02	1.68e+02	1.86e+02
Pu-239	C	5.67e+03	7.32e+03	7.26e+03	6.14e+03	7.42e+03	7.26e+03	7.64e+03	7.47e+03	3.11e+03	2.88e+03	4.33e+03	5.09e+03	5.16e+03	5.30e+03	5.27e+03	5.60e+03
	M	5.95e+03	7.87e+03	8.01e+03	6.60e+03	7.98e+03	7.68e+03	6.95e+03	6.82e+03	3.02e+03	2.83e+03	4.65e+03	5.08e+03	5.30e+03	5.47e+03	4.97e+03	5.32e+03
Pu-240	C	1.05e+03	2.05e+03	2.10e+03	1.37e+03	2.15e+03	2.10e+03	2.36e+03	2.31e+03	4.13e+02	3.25e+02	9.57e+02	1.41e+03	1.91e+03	2.06e+03	2.13e+03	2.19e+03
	M	1.12e+03	2.12e+03	2.26e+03	1.44e+03	2.37e+03	2.27e+03	2.57e+03	2.48e+03	4.22e+02	3.44e+02	1.03e+03	1.49e+03	2.10e+03	2.27e+03	2.32e+03	2.43e+03
Pu-241	C	6.40e+02	1.54e+03	1.57e+03	8.89e+02	1.63e+03	1.57e+03	1.83e+03	1.77e+03	1.17e+02	8.50e+01	3.92e+02	6.74e+02	9.34e+02	1.03e+03	1.07e+03	1.14e+03
	M	6.63e+02	1.54e+03	1.64e+03	9.15e+02	1.67e+03	1.58e+03	1.68e+03	1.62e+03	1.09e+02	8.28e+01	4.08e+02	6.51e+02	9.55e+02	1.06e+03	9.83e+02	1.08e+03
Pu-242	C	7.19e+01	3.58e+02	3.83e+02	1.34e+02	4.02e+02	3.85e+02	5.11e+02	4.92e+02	9.35e+00	5.46e+00	6.49e+01	1.62e+02	3.57e+02	4.38e+02	4.86e+02	5.03e+02
	M	8.03e+01	3.46e+02	3.98e+02	1.45e+02	4.22e+02	4.00e+02	5.52e+02	5.29e+02	9.49e+00	6.04e+00	7.39e+01	1.76e+02	4.08e+02	4.90e+02	5.34e+02	5.70e+02

Table 2-1. Calculated and Measured Isotopic Concentrations for the Selected Actinides (Continued)

		Trino Vercellese													Turkey Point				
Isotope	Axial Location (cm)	79.2	158.5	79.2	26.4	237.7	211.3	158.5	79.2	158.5	79.2	158.5	79.2	158.5	79.2	167.6	167.0	167.0	167.6	167.0
	Burnup (GWd/MTU)	12.04	15.38	15.90	11.53	12.86	20.60	23.72	24.30	23.87	24.55	23.93	24.36	24.33	24.31	30.72	30.51	31.56	31.26	31.31
	Enrichment (wt-% U-235)	3.90	3.13	3.13	3.13	3.13	3.13	3.13	3.13	3.13	3.13	3.13	3.13	3.13	3.13	2.56	2.56	2.56	2.56	2.56
	ALA	15.82	15.83	15.91	15.98	15.84	15.80	15.82	15.90	15.82	15.89	15.82	15.90	15.82	15.90	16.27	16.27	16.27	16.27	16.27
	Specific Power (MW/MTU)	15.42	19.69	20.36	14.76	10.74	17.21	19.82	20.30	19.94	20.51	19.99	20.35	20.33	20.31	32.24	32.01	33.12	32.80	32.85
U-234	C															1.38e+02	1.38e+02	1.36e+02	1.37e+02	1.36e+02
	M															1.32e+02	1.32e+02	1.23e+02	1.13e+02	1.32e+02
U-235	C	2.69e+04	1.77e+04	1.73e+04	2.04e+04	1.94e+04	1.46e+04	1.29e+04	1.25e+04	1.28e+04	1.24e+04	1.28e+04	1.25e+04	1.26e+04	1.25e+04	5.48e+03	5.55e+03	5.23e+03	5.32e+03	5.30e+03
	M	2.66e+04	1.73e+04	1.66e+04	2.02e+04	1.95e+04	1.44e+04	1.25e+04	1.24e+04	1.29e+04	1.22e+04	1.28e+04	1.23e+04	1.30e+04	1.23e+04	5.87e+03	5.68e+03	5.58e+03	5.51e+03	5.66e+03
U-236	C	2.46e+03	2.62e+03	2.68e+03	2.15e+03	2.33e+03	3.15e+03	3.41e+03	3.46e+03	3.42e+03	3.47e+03	3.43e+03	3.46e+03	3.46e+03	3.46e+03	3.35e+03	3.34e+03	3.38e+03	3.37e+03	3.37e+03
	M	2.74e+03	2.83e+03	2.74e+03	2.50e+03	2.45e+03	3.32e+03	3.61e+03	3.64e+03	3.52e+03	3.54e+03	3.75e+03	3.47e+03	3.47e+03	3.57e+03	3.25e+03	3.26e+03	3.17e+03	3.16e+03	3.25e+03
U-238	C	9.52e+05	9.56e+05	9.56e+05	9.59e+05	9.58e+05	9.51e+05	9.49e+05	9.49e+05	9.49e+05	9.48e+05	9.49e+05	9.49e+05	9.48e+05	9.49e+05	9.49e+05	9.49e+05	9.48e+05	9.49e+05	9.49e+05
	M	9.51e+05	9.56e+05	9.56e+05	9.60e+05	9.59e+05	9.52e+05	9.49e+05	9.49e+05	9.49e+05	9.48e+05	9.49e+05	9.48e+05	9.49e+05	9.47e+05	9.50e+05	9.51e+05	9.50e+05	9.50e+05	9.50e+05
Pu-238	C															1.47e+02	1.45e+02	1.56e+02	1.53e+02	1.53e+02
	M															1.37e+02	1.36e+02	1.43e+02	1.38e+02	1.37e+02
Pu-239	C	4.49e+03	5.30e+03	5.27e+03	4.40e+03	4.82e+03	6.01e+03	6.27e+03	6.17e+03	6.28e+03	6.19e+03	6.29e+03	6.17e+03	6.32e+03	6.17e+03	5.12e+03	5.12e+03	5.13e+03	5.13e+03	5.13e+03
	M	4.59e+03	5.27e+03	5.23e+03	4.42e+03	4.58e+03	5.76e+03	5.90e+03	6.07e+03	5.95e+03	5.98e+03	6.06e+03	6.00e+03	6.06e+03	5.97e+03	4.84e+03	4.84e+03	4.93e+03	4.94e+03	4.79e+03
Pu-240	C	6.55e+02	1.05e+03	1.09e+03	7.30e+02	8.48e+02	1.46e+03	1.68e+03	1.71e+03	1.69e+03	1.72e+03	1.69e+03	1.71e+03	1.72e+03	1.71e+03	2.15e+03	2.14e+03	2.19e+03	2.18e+03	2.18e+03
	M	7.17e+02	1.12e+03	1.14e+03	7.75e+02	8.40e+02	1.52e+03	1.76e+03	1.83e+03	1.76e+03	1.79e+03	1.79e+03	1.81e+03	1.77e+03	1.79e+03	2.27e+03	2.29e+03	2.30e+03	2.32e+03	2.28e+03
Pu-241	C	3.33e+02	6.36e+02	6.57e+02	3.70e+02	4.38e+02	9.33e+02	1.12e+03	1.13e+03	1.13e+03	1.15e+03	1.13e+03	1.14e+03	1.16e+03	1.14e+03	1.17e+03	1.16e+03	1.19e+03	1.19e+03	1.19e+03
	M	3.48e+02	6.14e+02	6.18e+02	3.69e+02	4.00e+02	8.85e+02	1.03e+03	1.06e+03	1.05e+03	1.06e+03	1.05e+03	1.06e+03	1.06e+03	1.06e+03	1.06e+03	1.07e+03	1.10e+03	1.12e+03	1.07e+03
Pu-242	C	2.48e+01	7.51e+01	8.17e+01	3.26e+01	4.40e+01	1.56e+02	2.20e+02	2.33e+02	2.23e+02	2.38e+02	2.25e+02	2.34e+02	2.34e+02	2.33e+02	5.14e+02	5.07e+02	5.44e+02	5.33e+02	5.35e+02
	M	3.14e+01	8.64e+01	9.49e+01	3.80e+01	4.60e+01	1.72e+02	2.44e+02	2.58e+02	2.40e+02	2.54e+02	2.47e+02	2.59e+02	2.44e+02	2.50e+02	5.02e+02	5.25e+02	5.48e+02	5.43e+02	5.24e+02

Table 2-1. Calculated and Measured Isotopic Concentrations for the Selected Actinides (Continued)

Isotope		Calvert Cliffs Unit 1 D047, MKP109 (ATM 104)			Calvert Cliffs Unit 1 D101, MLA098 (ATM-103)			Calvert Cliffs Unit 1 BT03, NBD107 (ATM-106)			H. B. Robinson B05, N-9 (ATM-101)				Obrigheim 170, 172, 176, 168, 171, 176					
	Axial Location (cm)*	13.2	27.7	165.2	9.1	24.5	161.9	11.3	19.9	161.2	11.0	26.0	199.0	226.0	N/A	N/A	N/A	N/A	N/A	N/A
	Burnup (GWd/MTU)	27.35	37.12	44.34	18.68	26.62	33.17	31.40	37.27	46.46	16.02	23.81	28.47	31.66	25.93	26.54	27.99	28.40	29.04	29.52
	Enrichment (wt-% U-235)	3.04	3.04	3.04	2.72	2.72	2.72	2.45	2.45	2.45	2.56	2.56	2.56	2.56	3.13	3.13	3.13	3.13	3.13	3.13
	ALA	16.40	16.37	16.28	16.46	16.42	16.33	16.28	16.26	16.18	16.46	16.41	16.27	16.23	16.31	16.31	16.30	16.30	16.30	16.29
	Specific Power (MW/MTU)	16.94	22.99	27.46	16.19	23.07	28.75	15.94	18.92	23.59	20.05	29.80	35.63	39.62	30.69	31.41	33.12	33.61	34.37	34.93
U-234	C	1.83e+02	1.58e+02	1.42e+02	1.82e+02	1.61e+02	1.46e+02	1.37e+02	1.26e+02	1.11e+02										
	M	1.82e+02	1.59e+02	1.36e+02	1.59e+02	1.37e+02	1.36e+02	1.74e+02	1.44e+02	8.50e+01										
U-235	C	9.08e+03	5.36e+03	3.63e+03	1.14e+04	7.53e+03	5.29e+03	4.38e+03	2.95e+03	1.58e+03	1.22e+04	8.26e+03	6.63e+03	5.66e+03	1.07e+04	1.04e+04	9.69e+03	9.51e+03	9.22e+03	9.01e+03
	M	9.61e+03	5.87e+03	4.02e+03	1.16e+04	7.87e+03	5.42e+03	4.38e+03	3.07e+03	1.60e+03	1.21e+04	8.18e+03	7.01e+03	5.51e+03	1.10e+04	1.06e+04	9.85e+03	9.68e+03	9.58e+03	9.18e+03
U-236	C	3.67e+03	4.12e+03	4.26e+03	2.82e+03	3.38e+03	3.65e+03	3.30e+03	3.43e+03	3.46e+03	2.45e+03	3.04e+03	3.27e+03	3.39e+03	3.64e+03	3.68e+03	3.78e+03	3.81e+03	3.84e+03	3.87e+03
	M	3.56e+03	4.00e+03	4.19e+03	2.84e+03	3.39e+03	3.70e+03	3.24e+03	3.44e+03	3.45e+03	2.48e+03	3.11e+03	3.20e+03	3.40e+03	3.59e+03	3.62e+03	3.70e+03	3.73e+03	3.75e+03	3.81e+03
U-238	C	9.50e+05	9.41e+05	9.34e+05	9.59e+05	9.52e+05	9.46e+05	9.50e+05	9.45e+05	9.36e+05	9.62e+05	9.55e+05	9.51e+05	9.48e+05						
	M	9.56e+05	9.45e+05	9.36e+05	9.70e+05	9.69e+05	9.55e+05	9.58e+05	9.57e+05	9.38e+05	9.61e+05	9.61e+05	9.46e+05	9.55e+05						
Pu-238	C	1.11e+02	2.13e+02	3.05e+02	4.53e+01	1.02e+02	1.67e+02	1.66e+02	2.26e+02	3.21e+02	3.23e+01	7.91e+01	1.20e+02	1.51e+02	8.60e+01	9.23e+01	1.04e+02	1.07e+02	1.12e+02	1.17e+02
	M	1.15e+02	2.14e+02	3.05e+02	5.50e+01	1.10e+02	1.68e+02	1.62e+02	2.21e+02	3.22e+02	3.21e+01	7.88e+01	1.29e+02	1.47e+02	8.01e+01	8.89e+01	9.48e+01	1.05e+02	1.01e+02	1.07e+02
Pu-239	C	4.86e+03	5.01e+03	5.17e+03	4.41e+03	4.79e+03	5.02e+03	4.40e+03	4.42e+03	4.50e+03	4.39e+03	4.88e+03	5.21e+03	5.34e+03	5.09e+03	5.11e+03	5.16e+03	5.18e+03	5.20e+03	5.22e+03
	M	4.84e+03	4.94e+03	4.94e+03	4.48e+03	4.82e+03	4.75e+03	4.33e+03	4.35e+03	4.27e+03	4.13e+03	4.56e+03	4.98e+03	4.76e+03	4.81e+03	4.71e+03	4.93e+03	5.01e+03	4.96e+03	4.94e+03
Pu-240	C	1.83e+03	2.34e+03	2.64e+03	1.31e+03	1.85e+03	2.21e+03	2.19e+03	2.43e+03	2.71e+03	1.21e+03	1.81e+03	2.11e+03	2.29e+03	1.70e+03	1.75e+03	1.84e+03	1.86e+03	1.89e+03	1.92e+03
	M	1.95e+03	2.54e+03	2.88e+03	1.41e+03	2.00e+03	2.39e+03	2.34e+03	2.63e+03	2.95e+03	1.24e+03	1.89e+03	2.23e+03	2.41e+03	1.80e+03	1.83e+03	1.92e+03	2.02e+03	2.00e+03	2.04e+03
Pu-241	C	8.04e+02	1.06e+03	1.21e+03	5.11e+02	7.91e+02	9.81e+02	8.24e+02	9.19e+02	1.03e+03	3.63e+02	6.04e+02	7.74e+02	8.53e+02	1.04e+03	1.07e+03	1.13e+03	1.15e+03	1.18e+03	1.20e+03
	M	7.73e+02	1.02e+03	1.16e+03	5.15e+02	7.74e+02	9.22e+02	8.24e+02	9.22e+02	1.01e+03	3.45e+02	5.72e+02	7.73e+02	7.85e+02	9.78e+02	9.78e+02	1.06e+03	1.10e+03	1.11e+03	1.13e+03
Pu-242	C	3.14e+02	6.30e+02	8.91e+02	1.41e+02	3.45e+02	5.64e+02	5.69e+02	8.02e+02	1.17e+03					2.72e+02	2.86e+02	3.27e+02	3.39e+02	3.58e+02	3.71e+02
	M	3.28e+02	6.53e+02	9.53e+02	1.58e+02	3.74e+02	6.21e+02	6.20e+02	8.80e+02	1.33e+03					3.12e+02	3.28e+02	3.72e+02	4.07e+02	4.05e+02	4.38e+02

C = Calculated value
M = Measured value
*Height of sample above bottom of fuel
N/A = Not Applicable. Isotopic measurements were performed by assaying full length halves of fuel.

provides the range of values for spent fuel assembly dimensions and composition from which the 54 chemical assays were taken and those assemblies anticipated for burnup credit.

Table 2-2 Comparison of Physical Parameters Between Benchmark and Anticipated Burnup Credit Assemblies

Key Parameters	Anticipated Burnup Credit Spent Fuel PWR Assemblies	54 Benchmark Samples
Assembly Characteristics		
Type	All PWR types	CE 14x14 WE 17x18, 15x15, 14x15 Siemens 14x14
Fuel Rod Characteristics		
Pitch (cm)	1.07 - 1.47	1.07 - 1.47
Outside Diameter (cm)	0.79 - 1.12	0.86 - 1.12
Initial Gap (cm)	0.005 - 0.017	0.005 - 0.015
Pellet Diameter (cm)	0.71 - 0.99	0.75 - 0.96
Pellet Density (g/cm^3)	10.0 - 10.4	10.08 - 10.57
Moderator/Fuel Volume	1.3 - 1.9	1.3 - 1.7
Clad Material	Zircaloy Stainless Steel	Zircaloy Stainless Steel
Cooling Time (years)	5 to 200	0 - 10*

*Time of reported measurements

The chemical assays are used to validate the production and destruction of nine actinide isotopes during incore operation. To explain the calculational process, the following differential equations are provided for each isotope. Some simplification is taken (decay with long half lives and lesser precursors are ignored); however, in no case does the simplification represent more than a 1% effect.

$$dN^{U234}/dt = - N^{U234}\sigma_a{}^{U234}\phi$$

$$dN^{U235}/dt = (N^{U234}\sigma_c{}^{U234} - N^{U235}\sigma_a{}^{U235})\phi$$

$$dN^{U238}/dt = - N^{U238}\sigma_a{}^{U238}\phi$$

$$dN^{Pu238}/dt = \lambda^{Np238}N^{Np238} - N^{Pu238}\sigma_a{}^{Pu238}\phi \qquad \text{See Note 1}$$

$$dN^{Pu239}/dt = (N^{U238}\sigma_c{}^{U238} + N^{Pu238}\sigma_c{}^{Pu238} - N^{Pu239}\sigma_a{}^{Pu239})\phi \qquad \text{See Note 2}$$

$$dN^{Pu240}/dt = (N^{Pu239}\sigma_c{}^{Pu239} - N^{Pu240}\sigma_a{}^{Pu240})\phi$$

$$dN^{Pu241}/dt = (N^{Pu240}\sigma_c{}^{Pu240} - N^{Pu241}\sigma_a{}^{Pu241})\phi - \lambda^{Pu241}N^{Pu241}$$

$$dN^{Pu242}/dt = (N^{Pu241}\sigma_c{}^{Pu241} - N^{Pu242}\sigma_a{}^{Pu242})\phi$$

where:

N	is atom density which is a function of time
σ_c and σ_a	are the one-group capture and absorption cross section, respectively
ϕ	is the flux, and
λ	decay constant.

Note 1: The Pu-238 equation depends on Np-238 which is first introduced in this equation. The Np-238 is produced through captures in U-236, followed by Np-237. It also requires the decay of U-237 (with its half life of 6.75 days) to Np-237. The additional equations needed follow:

$$dN^{U237}/dt = (N^{U236}\sigma_c{}^{U236} - N^{U237}\sigma_a{}^{U237})\phi - \lambda^{U237}N^{U237}$$
$$dN^{Np237}/dt = \lambda^{U237}N^{U237} - N^{Np237}\sigma_a{}^{Np237}\phi$$
$$dN^{Np238}/dt = (N^{Np237}\sigma_c{}^{Np237} - N^{Np238}\sigma_a{}^{Np238})\phi - \lambda^{Np238}N^{Np238}$$

Note 2: The Np-239 step has been ignored. It has a half life of 2.355 days and a small absorption cross section, so it can be assumed that 100% of the captures in U-238 become Pu-239. ORIGEN tracks this isotope so a decay of about 10 days after shutdown is necessary to allow the conversion of the inventory of Np-239 to Pu-239.

As an example, the solution of the U-238 differential equation is:

$$N^{238}(t) = N^{238}(0) \exp(-\sigma_a{}^{U238}\phi t) \qquad \text{Eq. 2-1}$$

This solution is written in a constant flux form. The non-constant flux version would only require changing the exponent to an integral as a function of time. Equation 2-1 can be modified to a form that includes burnup using the following relationship:

$$\text{Burnup} = \kappa \Sigma_f \phi \, t / D \qquad \text{Eq. 2-2}$$

where:

- κ is the energy per fission
- Σ_f is the macroscopic fission cross section
- t is the irradiation time
- D is the heavy metal density

Solving equation 2-2 for the flux times time and placing it into equation 2-1 results in:

$$N^{238}(t) = N^{238}(0) \exp(-\sigma_a{}^{U238} * \text{Burnup} * D / \kappa\Sigma_f) \qquad \text{Eq. 2-3}$$

Equation 2-3 is for U-238, but it demonstrates the key characteristics of the depletion equations.

As can be seen in equation 2-3, burnup represents the time in the depletion equations. With no burnup, the initial condition would result. If the depletion analysis contains biases, it would be expected that the biases would increase with burnup and would go to zero with zero burnup. Any error in a cross section is expected to demonstrate itself in a trend with burnup. A trend with burnup will expose problems with both the direct capture and absorption cross sections from the library (e.g., 27BURNUPLIB) as well as the processing technique. Thus, analysis should be performed to seek a trend on burnup.

All of the depletion equations depend on one-group cross sections. In order to get the correct one-group cross section, the energy spectrum of the flux must be accurate. Some of the items that affect the neutron energy spectrum in a PWR are:

- Assembly design (there is a small variation of the hydrogen to uranium ratios between assembly designs)
- Moderator density (temperature is the measured parameter)
- Soluble boron concentration
- Presence of burnable absorbers

- Burnup
- Specific Power
- Enrichment.

Since the collapse of the cross sections depends on the spectrum, trends on spectrum should be sought. The magnitude of the effect of a spectral error will depend on the quantity of burnup. Consequently, the trend sought will actually be the product of a spectral index and the burnup.

In Equation 2-3, the macroscopic fission cross section appears in the exponential term. With higher enrichments, the macroscopic fission cross section increases. A trend on enrichment should be sought in order to detect any deviations in isotopic production caused by errors in the fission cross section. Again, the observed quantity of this error will increase with burnup so that the trending analysis will seek a trend on the product of burnup times enrichment. The uncertainty in isotopic concentrations due to the dependency on enrichment should be small. This is because the macroscopic fission cross section is very important to reactivity control of reactors. For example, a 1% error in Σ_f would produce a 1% error in k_{eff}, which would be large in criticality analysis. In fact, commercial reactors are required to shut down if there is a reactivity anomaly of 1%. Since the impact of enrichment on isotopic concentration uncertainty is small, it is appropriate to allow extrapolation of any trend observed to cover the range of commercial fuel.

Knowing the specific power and burnup sets the irradiation time. For Pu-238 and Pu-241, the isotopic concentration depends on the competition of absorption and decay. A trend on specific power should be sought in order to determine any bias in the decay constant relative to the capture cross section. The magnitude of any trend on specific power depends on the burnup. Therefore, a search for a trend on Pu-238 and Pu-241 as a function of burnup times specific power should be performed. It is inappropriate to seek a trend on specific power for any of the other actinides (U-234, U-235, U-238, Pu-239, Pu-240, or Pu-242) since radioactive decay is insignificant.

Based on the above discussion, it is apparent that the range of applicability can be described by four parameters: burnup, spectrum, enrichment, and specific power. These four parameters cover the major independent ways of creating errors in isotopic concentrations. Table 2-3 represents the range of applicability that can be supported by the data. Note that for some parameters, the range of applicability is greater than the range of the data from the benchmarking experiments. In these cases, the data range is extrapolated via a trending analysis as permitted by ANSI/ANS-8.1. In the case of specific power, the extrapolation to 60 MW/MTU is acceptable, since specific power only effects Pu-238 and Pu-241, and as shown in DOE/RW-0495, Figure 4-7, the effect is small over a very large specific power range.[2-13] In Table 2-3, the spectral index is given as the Average Lethargy of Absorption (ALA). Any spectral index is acceptable, but this is the recommended parameter. Equation 2-4 below defines the ALA.

$$ALA = \frac{\sum_{i=1}^{NG} \Phi_i \Sigma_i u_i}{\sum_{i=1}^{NG} \Phi_i \Sigma_i} \qquad \text{Eq. 2-4}$$

where,

u_i = average lethargy for energy group i

Φ_i = the flux for the i^{th} energy group

Σ_i = the absorption cross section for the i^{th} energy group.

NG = the number of energy groups

The ALA values calculated using Equation 2-4 for the 54 benchmarks range between 15.6 and 16.5. The data points around 15.6 belong to Yankee Rowe samples, which indicate a hard neutron spectrum. This is confirmed by the fact that the samples come from rods with smallest pitch and stainless steel cladding. All the data between 16 and 16.5 belong to samples which came from typical commercial PWRs (Calvert Cliffs, H. B. Robinson, Obrigheim, etc.). Therefore, the 54 benchmarks considered for isotopic validation cover a wide range of neutron spectra, and even cover fuel with hard spectra (Yankee Rowe and Trino Vercellese) which are not usually seen among typical PWRs.

Table 2-3. Range of Applicability Matrix for Isotopic Validation

Key Parameters	Range of 54 Benchmark Experiments	Range of Applicability
Burnup (GWd/MTU)	6.9 - 46.46	0 - 40
Spectral Index[1]	15.6 - 16.5	15 - 17
Initial Enrichment (Wt. % U-235)	2.45 - 3.90	1.5 - 4.05
Specific Power(MW/MTU)	10.7 - 39.6	0 - 60[2]

[1]The spectral index is the Average Lethargy of Absorption with a reference energy of 20 Mev.

[2]For the depletion analysis, a constant specific power of 60 MW/MTU is used for conservatism.

2.2.3 Qualification of Isotopic Measurement Data

The radiochemical analyses of spent fuel isotopic composition samples from the Calvert Cliffs and H. B. Robinson reactors were analyzed by the MCC at PNL. The MCC is responsible for providing spent fuel Approved Testing Materials (ATMs) for radiochemical measurements

conducted by PNL for the Department of Energy's Office of Civilian Radioactive Waste Management. The activities at PNL were performed according to QA procedures developed using the QA program, which included ASME/ANSI NQA-1 requirements.

Although the programs under which measurements on the samples from Obrigheim, Trino Vercellese, Yankee Rowe, Turkey Point, and Mihama have not been performed under the DOE OCRWM QA program, the production and handling of radioactive materials, especially fissile materials, is heavily controlled by both national and international regulatory bodies. Detailed procedures and documentation are required for activities utilizing these materials. One of the primary purposes of a formal QA program is to establish written policies and procedures to ensure good scientific principles are utilized. The nature of material used in these studies (solutions containing fissionable materials) provides assurance that stringent requirements for formal procedures and documentation were imposed on the experiments. Further assurance of the technical quality of the test results on most of the samples is provided through multiple radiochemical analyses performed by independent laboratories. For example, fuel samples from the Obrigheim PWR in Germany were analyzed independently by four European laboratories; European Institute for Transuranic Elements, Institute for Radiochemistry, Karlsruhe Reprocessing Plant, and the International Atomic Energy Agency. The isotopic measurements for Trino Vercellese spent fuel were conducted by Ispara and Karlsruhe. The Yankee Rowe samples were analyzed by TRACELAB, GE-Vallecitos, and New Brunswick laboratories.

2.2.4 Summary of Approval Requests on Isotopic Validation Data

Section 2.2.1 described the results of the measurements for the selected actinides to be used in validation of depletion computer codes for burnup credit analysis. Section 2.2.2 provided the justification for the acceptability of the measurement data in terms of the range of applicability for the validation purposes. Based on the previous discussions, acceptance of the measurement data, summarized in Table 2-1, for the selected actinides is requested.

2.3 ISOTOPIC CALCULATIONAL BIAS AND UNCERTAINTIES

Any validated depletion computer code can be used to perform isotopic depletion/generation for burnup credit. In order to validate the code, each of the 54 cases shown in Table 2-1 must be analyzed. The analysis technique must be the same as that which will be used for the burnup credit package analysis.

The requirement of using the same technique for both validation and package analysis is not expected to be very restrictive; however, an example violation may help in understanding this requirement. In any depletion code, an approximation of the energy distribution of the flux must be made. In a point depletion code, such as SAS2H sequence, the geometric data that can be

handled is limited; therefore, a general algorithm is set up and validated. The algorithm selected for assembly models performed for this study is:

1) Divide the assembly cross sectional area by the number of removable burnable absorber rods (or by the number of guide tubes plus one for the instrumentation tube if there are no burnable absorbers).
2) Model the assembly with concentric rings to preserve the volume of each component of that fraction of the assembly.
3) The central rings will be related to the cell associated with a burnable absorber (or a guide tube). This would then be surrounded by the appropriate quantity of homogenized fuel cells. The next ring will contain the appropriate volume of guide tube cell material. The final ring will contain the assembly gap material.

The resulting energy spectrum of the flux is dependent on this algorithm. Although a number of the chemical assay points are taken from positions in the assembly that are not well represented by this algorithm, it is inappropriate to use a different algorithm for the assembly and point calculation. Hence, a violation of the modeling requirement would result if the analyst attempts to model a chemical assay by rings around the particular assayed pellet. If a 2D computer code such as CASMO, PHOENIX, or HELIOS is used, data can be obtained from the appropriate point in the assembly without modifying the assembly homogenization algorithm. With that type of code, the actual energy spectrum associated with the chemical assay point can be used in the validation process.

After the isotopic inventory for the 54 benchmarks is calculated, a statistical method is used to determine the bias, along with the uncertainties, in terms of a set of correction factors. These correction factors are then used to adjust the future isotopic values calculated by the code, for which the correction factors were developed. The following subsections present the statistical method to be used in developing the correction factors.

2.3.1 Definition of the Bias

Given the calculated and measured values for isotopic concentrations, this section defines a bias to be used for adjusting the calculated isotopic concentrations to the best estimate isotopic concentrations. The bias approach selected for isotopics is a multiplicative bias, x , which is the ratio of measured to calculated values. The bias and its associated uncertainty are then used to determine a set of correction factors for adjusting the future isotopic values calculated by the particular code.

To use the bias in determining the correction factors by which the calculated isotopic values can be simply multiplied, the bias should be calculated in terms of the ratio between the measured and the calculated values:

$$x = \frac{M}{C}$$ Eq. 2-5

where:

x = multiplicative bias

M = measured isotopic concentration value

C = calculated isotopic concentration value.

If multiple measurements were performed on a sample, the average of all the measurements was used for M. For example, for Obrigheim, Yankee Rowe, and Trino Vercellese, average values of the multiple independent measurements were used for M.

The relationship between the measured-to-calculated ratio and the input parameters (i.e., trending parameters) is assumed to be as follows:

$$x_{fit} = (M/C)_{fit} = 1.0 + b_1*B + b_2*B*S + b_3*B*E + b_4*B*P$$ Eq. 2-6

where:

x_{fit} = predicted bias as a function of input parameters
B = burnup (GWd/MTU)
b_1 = slope for burnup
S = a spectral index (ALA)
b_2 = slope for product of burnup*spectral index
E = initial enrichment (wt.% initial U-235)
b_3 = slope for product of burnup*initial enrichment
P = specific power (MW/MTU)
b_4 = slope for product of burnup*specific power

As seen, the burnup variable appears in each of the terms on the right side of the equation. This is because the amount of change in x_{fit} due to spectrum, enrichment, and specific power related problems is proportional to burnup. The x_{fit} value at zero burnup is one because the calculated value becomes the initial condition measured value if there is no burnup.

2.3.2 Regression Analysis

The slopes for each of the variables, b_1, b_2, b_3, b_4, are determined one by one sequentially, and a test is performed to show whether the slopes are significant or not. This is being done since the parameters (burnup, spectrum, enrichment, and specific power) are dependent. For example, the

burnup, enrichment, and specific power all effect the spectral index. It is desirable to lump all the effects due to increased burnup with the burnup term. It is then desirable to find trends on spectrum that did not correlate with burnup. Finally, trends are not expected on enrichment or specific power and thus are held to the end to prevent identification of spurious trends.

The procedure starts by determining b_1 by assuming x_{fit} to be a function of burnup only.

$$x_{fit} = 1 + b_1 * B \qquad \text{Eq. 2-7}$$

Using the single-parameter regression model shown by Eq. 2-7, the slope b_1 is determined. The value for b_1 is calculated by minimizing the following equation:[2-14]

$$SS_R = \sum_{i=1}^{n} (x_i - 1 - b_1 * B_i)^2 \qquad \text{Eq. 2-8}$$

where,

- n = number of data points
- SS_R = the sum of the squares of differences
- x_i = measured-to-calculate value for the i^{th} data point
- B_i = burnup value for the i^{th} data point

Setting the derivative of SS_R with respect to b_1 to zero, and solving for b_1 results in the following equation:

$$b_1 = \frac{\sum_{i=1}^{n} B_i * x_i - \sum_{i=1}^{n} B_i}{\sum_{i=1}^{n} B_i^2} \qquad \text{Eq. 2-9}$$

A test is then performed to determine if the value for parameter b_1 is significant. The null hypothesis in this test is $b_1 = 0$.

The test statistic is:[2-9]

$$v = |b_1| \sqrt{\frac{(n-1)\sum_{i=1}^{n} B_i^2}{SS_R}} \qquad \text{Eq. 2-10}$$

After calculation of the test statistic, it is compared to the Student's t-value (found in Appendix A.3 of Reference 2-15), for the particular sample size (i.e., n-1) and level of significance. The null hypothesis is justified if the calculated test statistic is less than the Student's t-value. The level of significance for this test has been selected at 95% confidence. Therefore, the trend will be rejected unless there is 95% confidence that the slope is not zero.

If the test indicates the value for b_1 is not significant, the burnup term is eliminated (i.e., $b_1 = 0$) and a regression with respect to burnup*spectral index is performed with the intercept value still being one. The value for b_2 is then calculated similar to Equation 2-9. This process is repeated and each time a test is performed to determine whether the value for the respective slope is significant or not.

If the trending test indicates the value for the slope associated with burnup is significant, the burnup term is kept in the equation, and Equation 2-7 for burnup*spectral index becomes as follows:

$$x_{fit} = 1 + b_1*B + b_2*B*S \qquad \text{Eq. 2-11}$$

The regression procedure, as before, calls for minimizing the following equation:

$$SS_R=\sum_{i=1}^{n} (x_i-1-b_1*B_i-b_2*B_i*S_i)^2 \qquad \text{Eq. 2-12}$$

Taking the partial derivative of Eq. 2-12 with respect to b_2 and setting it equal to zero, yields the following equation,

$$b_2=\frac{\sum_{i=1}^{n} B_i*S_i*x_i-\sum_{i=1}^{n} B_i*S_i-b_1*\sum_{i=1}^{n} B_i*S_i*B_i}{\sum_{i=1}^{n} (B_i*S_i)^2} \qquad \text{Eq. 2-13}$$

As before, the trending test is applied to b_2. The resultant trending test formula for b_2 is similar to Equation 2-10 for testing b_1, except b_1 and B_i are replaced with b_2 and B_i*S_i.

Again, if the trending test indicates the value for the slope associated with burnup*spectral index is significant, the burnup*spectral index term is kept in the equation, and Equation 2-11 becomes as follows:

$$x_{fit} = 1 + b_1*B + b_2*B*S + b_3*B*E \qquad \text{Eq. 2-14}$$

As before, in order to determine b_3, the sum of the squares of the residuals which is:

$$SS_R=\sum_{i=1}^{n} (x_i-1-b_1*B_i-b_2*B_i*S_i-b_3*B_i*E_i)^2 \qquad \text{Eq. 2-15}$$

is minimized by setting the partial derivative of Equation 2-15 with respect to b_3 to zero. This results in the following equation for b_3:

$$b_3 = \frac{\sum_{i=1}^{n} B_i * E_i * x_i - \sum_{i=1}^{n} B_i * E_i - b_1 * \sum_{i=1}^{n} B_i * E_i * B_i - b_2 * \sum_{i=1}^{n} B_i * E_i * B_i * S_i}{\sum_{i=1}^{n} (B_i * E_i)^2} \qquad \text{Eq. 2-16}$$

The trending test formula for b_3 is again similar to Equation 2-10 for testing b_1, but with replacement of the appropriate parameters, which in this case are b_3 and B_i*E_i. If the test indicates the value of b_3 is significant, Equation 2-6 becomes the final equation with b_1, b_2, and b_3 values given by Equations 2-9, 2-13, and 2-16. However, for Pu-238 and Pu-241, trending analysis with respect to a fourth parameter, burnup*specific power is also performed using a similar procedure to that above. The Equation for b_4 can be determined by inspecting Equations 2-9, 2-13, and 2-16. The test statistic for testing b_4 is similar to Equation 2-10, but with b_4 and B_i*P_i as the independent variables.

2.3.3 Correction Factors

Having established the trends associated with the isotopic data, the correction factor for each isotope can now be determined. The correction factor for each isotope is determined to assure that the calculated isotopic concentration is conservative. This implies adding an appropriate uncertainty to the calculated fissile isotopes and subtracting the uncertainty from the calculated absorbers. The appropriate uncertainty is found using the prediction interval technique. This technique establishes an interval around the mean prediction in which there is 95% confidence that the next observation will be within the interval. For this application, only one side of the interval is of interest. Therefore, the uncertainty is established in a way that there is a 95% confidence that the next observation will be above (absorbers) or below (fissile isotopes) the corrected isotopic concentration.

The correction factor for those isotopes which do not exhibit any trends is:[2-16]

$$f = 1.0 \pm \sigma t_{95,n-1}\sqrt{1+1/n} \qquad \text{Eq. 2-17}$$

where,

$$\sigma = \sqrt{\frac{1}{n}\sum_{i=1}^{n} (x_i - 1)^2} \qquad \text{Eq. 2-18}$$

and $t_{95,n-1}$ is the Student-t value for 95% confident with n-1 data points. As indicated in Equation 2-18, the uncertainty around one is used in the absence of any trends. The sign between the bias and the total uncertainty depends on the type of isotope. For positive worth isotopes, the correction factor is calculated by adding 1.0 to the total uncertainty as follow:

$$f_{buc} = f_{pos} = 1.0 + \sigma t_{95,n-1}\sqrt{1+1/n} \qquad \text{Eq. 2-19}$$

The correction factors for negative worth isotopes (neutron absorbers) are calculated in a similar but converse manner.

$$f_{buc} = f_{neg} = 1.0 - \sigma t_{95,n-1}\sqrt{1+1/n} \qquad \text{Eq. 2-20}$$

The generic correction factor formula for isotopes which exhibit trends with respect to one or more of the four parameters (burnup, burnup*spectrum, burnup*initial enrichment, and burnup*specific power) is also determined based on a prediction interval. Therefore, the 95%-confidence correction factor is:

$$f = 1.0 + \sum_{j=1}^{m} b_j * H_j \pm t_{95,n-m}\sqrt{(1+\sum_{j=1}^{m}\frac{H_j^2}{\sum_{i=1}^{n} h_{ji}^2})\frac{SS_R}{n-m}} \qquad \text{Eq. 2-21}$$

where:

m = number of parameters (1,2,3, or 4) against which the specific isotope exhibited trends

b_j = slope for the trending parameter j (burnup, burnup*spectral index, burnup*initial enrichment, or burnup*specific power)

H_j = the value of trending parameter variable j

h_{ji} = the value for trending parameter j for the ith sample (The h_{ji} values are predetermined by the sample set.)

$$SS_R = \sum_{i=1}^{n} (x_i - 1 - \sum_{j=1}^{m} b_j * h_{ji})^2$$

n = number of data points

Similar to the non-trended case, the sign between the bias and the total uncertainty depends on the type of isotope. Furthermore, a level of conservatism is added to this process for fissile isotopes by ignoring the correction factor when its value is below unity. Hence, for a positive worth isotope whose calculated isotopic concentration is greater than the measured value, the calculated value is not lowered (i.e., f=1.0). This would ensure that the maximum amount of positive worth isotopes is included in the criticality calculations. Based on this conservatism, the correction factor for positive worth isotopes which exhibit trends with respect to one or more of the four parameters is:

$$f_{buc}=f_{pos}=\max\ [1.0+\sum_{j=1}^{m} b_j*H_j+t_{95,n-m}\sqrt{(1+\sum_{j=1}^{m}\frac{H_j^2}{\sum_{i=1}^{n} h_{ji}^2})\frac{SS_R}{n-m}},\ 1.0] \qquad \text{Eq. 2-22}$$

The correction factors for negative worth and trended isotopes are calculated by subtracting the total uncertainty from the bias. The conservative assumption on disallowing any compensating effects is considered by setting the correction factor equal to unity for non-fissile actinides if the calculated correction factor is greater than one. This approach ensures that the future calculated inventory for negative worth isotopes is not increased if the chemical assay indicated a higher measured value than the corresponding calculated value. Therefore, the correction factor for negative worth isotopes which exhibit trends with respect to one or more of the four parameters is:

$$f_{buc}=f_{neg}=\min\ [1.0+\sum_{j=1}^{m} b_j*H_j-t_{95,n-m}\sqrt{(1+\sum_{j=1}^{m}\frac{H_j^2}{\sum_{i=1}^{n} h_{ji}^2})\frac{SS_R}{n-m}},\ 1.0] \qquad \text{Eq. 2-23}$$

2.3.4 Summary of Approval Requests for Isotopic Calculational Methodology

Section 2.3 described methodology requirements for performing fuel depletion calculations and presented a statistical approach for calculating biases, uncertainties, and correction factors, based on calculated and measured isotopic values that can be used to bias future calculated isotopic values. Section 2.3 of this topical report seeks the NRC acceptance of the methodology for fuel depletion calculation and the proposed statistical approach in calculating biases, uncertainties, and correction factors.

2.4 DEMONSTRATION WITH SCALE 4.2 AND 27BURNUPLIB CROSS SECTIONS

The computational tool used to demonstrate the isotopic validation methodology, described in Sections 2.1 through 2.3, is the SAS2H calculational sequence from the SCALE 4.2 computer code package with the 27BURNUPLIB cross section library.[2-17] SAS2H/27BURNUPLIB invokes a series of cross section processing codes and a 1-D transport cell model that allows problem-specific (assembly type including water holes) cross sections to be used as a function of burnup. The core of the calculational sequence is the ORIGEN-S point depletion/decay code.

ORIGEN-S requires nuclide data such as cross section data, fission product yields, decay data, and branching fractions (the probability associated with a particular mode of decay). ORIGEN-S also requires system data such as initial fuel composition, fuel geometry, and the operating history of the fuel (e.g., specific power, exposure time, and down time). Nuclide data are supplied to ORIGEN-S by libraries within the SCALE system, while the system data are problem specific and user specified. As the calculation proceeds through the exposure history, cross section data are updated by the 1-D transport code based on revised (as calculated by ORIGEN-S) isotopic concentrations to capture the effects of shifts in the energy spectrum. The output of such a calculation provides the calculated isotopic concentration for user-specified nuclides. SAS2H modeling details for each experiment are described in References 2-9, 2-10, and 2-11. Table 2-1 provides the calculated values for each of the measured isotopic samples.

Based on the methodology described in Sections 2.3.1 through 2.3.3 and the data provided in Table 2-1, trending analyses were performed for the eight isotopes with respect to burnup, burnup*spectrum, and burnup*enrichment.[2-9] A trending test with respect to burnup*specific power was performed only for Pu-238 and Pu-241. The results of trending analyses for U-234, U-238, and Pu-238 did not indicate any trends with any of the parameters. However, U-235, Pu-239, Pu-240, Pu-241, and Pu-242 exhibited trends with respect to burnup only. Since the Am-241 correction factor is determined based on Pu-241, a correction factor with the same burnup trend is assigned to Am-241 (obviously, the sign for the uncertainty term changes). The correction factor equations for those parameters which exhibited trends with respect to burnup (in GWd/MTU) is provided in the following equations.

$$f_{buc}^{U\text{-}235} = \max\left[1.0+0.00105*B+1.6741\sqrt{(1.0+\frac{B^2}{41100})\frac{0.050}{54-1}},\ 1.0\right]$$

$$f_{buc}^{Pu\text{-}239} = \max\left[1.0-0.000852*B+1.6741\sqrt{(1.0+\frac{B^2}{41100})\frac{0.111}{54-1}},\ 1.0\right]$$

$$f_{buc}^{Pu\text{-}240} = \min\left[1.0+0.00231*B-1.6741\sqrt{(1.0+\frac{B^2}{41100})\frac{0.025}{54-1}},\ 1.0\right]$$

$$f_{buc}^{Pu\text{-}241} = \max\left[1.0-0.00142*B+1.6741\sqrt{(1.0+\frac{B^2}{41100})\frac{0.094}{54-1}},\ 1.0\right]$$

$$f_{buc}^{Pu-242} = \min\left[1.0+0.00300*B-1.6766\sqrt{(1.0+\frac{B^2}{38500})\frac{0.247}{50-1}},\ 1.0\right]$$

$$f_{buc}^{Am-241} = \min\left[1.0-0.00142*B-1.6741\sqrt{(1.0+\frac{B^2}{41100})\frac{0.094}{54-1}},\ 1.0\right]$$

Table 2-4 provides the mean biases, uncertainties, and the correction factors (f_{buc}) for each selected actinide. For those isotopes that exhibit a trend with burnup, the values in Table 2-4 are given for 30 GWd/MTU for demonstration purposes. The f_{buc} values are the only numbers that should be used in adjusting the calculated values for selected actinides, representing spent fuel composition, when performing burnup credit criticality analyses. The adjustment is performed by multiplying the number density for each of the selected actinide isotopes by the corresponding correction factor. This table is valid only with use of the SAS2H/27BURNUPLIB of SCALE 4.2 computer code system. A similar set of correction factors should be generated if another code system or cross section set is selected.

2.5 SUMMARY OF ISOTOPIC VALIDATION

In this chapter, three main components of isotopic validation were discussed: 1) experimental data, 2) validation methodology, and 3) the demonstration of the validation of SAS2H/27BURNUPLIB of the SCALE 4.2 computer code package. It is the purpose of this chapter to seek the acceptance of the first two of these three components of the isotopic validation.

The experimental data component presented in Section 2.2 was developed by performing chemical assays on selected rods from many spent fuel assemblies. The data cover a wide range of the parameters important to isotopic concentrations during fuel depletion in reactors. The validation methodology consists of best-estimate analysis for determining isotopic concentrations computationally, and the statistical approach in determining biases, uncertainties, and correction factors. The above validation methodology was demonstrated using the experimental data and SAS2H/27BURNUPLIB of the SCALE 4.2 computer code package.

Therefore, this section seeks NRC acceptance of the following:

- The PWR fuel post irradiation examination assay data selected in Table 2-1 for isotopic inventory bias and uncertainty determination are sufficient for validating the selected actinide composition in spent fuel

- The statistical procedure proposed for establishing isotope-specific biases and correction factors is a conservative method used to account for isotopic concentration changes during burnup.

Table 2-4. Bias, Uncertainty, and Isotopic Correction Factors for Burnup Credit Nuclides for Use with SCALE 4.2 and 27BURNUPLIB Analyses

Isotope	n	Mean Bias	Uncertainty	f	f_{buc}
U-234	25	1.000	0.182	0.818	0.818
*U-235	54	1.025	0.052	1.084	1.084
U-238	48	1.000	0.009	0.991	0.991
Pu-238	40	1.000	0.132	0.868	0.868
*Pu-239	54	0.979	0.078	1.052	1.052
*Pu-240	54	1.063	0.037	1.033	1.000
*Pu-241	54	0.960	0.071	1.028	1.028
*Pu-242	50	1.092	0.120	0.969	0.969
*Am-241**	N/A	0.960	0.071	0.886	0.886

*Evaluated at 30 GWd/MTU for demonstration purposes.

**Since the vast majority of Am-241 is created after shutdown by Pu-241, Am-241 was biased based on Pu-241.

Note: The mean bias for isotopes without a trend is 1.0 by definition. The mean of the samples for U-234, U-238, and Pu-238 are 0.973, 1.001, 1.005, respectively.

3. CRITICALITY VALIDATION

The criticality validation for spent fuel packages using burnup credit is very similar to that required for the fresh fuel assumption. The approach documented in "*Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages* (NUREG/CR-5661)"[3-1] and "*Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages* (NUREG/CR-6361)" [3-2] applies in general to burnup credit. This chapter will address only those issues on criticality validation not covered in the above referenced NUREG documents.

Chapter 5 of NUREG/CR-5661 is entitled "Validation of Calculational Method." It reviews the selection of critical experiments, establishment of bias and uncertainty, establishment of range of applicability, and finally, establishment of acceptance criteria. All of these steps must be performed whether or not burnup credit is utilized. This chapter will follow the same steps and cite additional requirements for actinide-only burnup credit.

3.1 SELECTION OF CRITICAL EXPERIMENTS

The NUREG/CR-5661 states that one selects critical experiments "that have parameters (e.g., materials, geometry, etc.) that are characteristic of the package design." This is still important for burnup credit designs and is part of the process for the lower enrichments where the "fresh fuel assumption" is still applied. The NUREG/CR-5661 then cites NUREG/CR-6361, which lists 180 UO_2 critical experiments for this purpose. For actinide-only burnup credit there is no change in the requirements for the selection of UO_2 critical experiments. This approach covers U-234, U-235, and U-238 but does not address the Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, or Am-241. To cover these isotopes, MOX critical experiments are used. Since SNF has a lower plutonium content than MOX fuel, criticality conditions of SNF are bounded by UO_2 experiments and MOX experiments. MOX experiments generally are simple lattice experiments and do not cover "characteristics of the package design." Therefore, a selection of MOX experiments that cover all designs is made in this topical report. The Table 3.1 lists a set of MOX experiments that should be used for criticality validation. The following subsections describe some of the features of the MOX critical experiments selected.

3.1.1 EPRI Clean Critical Experiment Benchmarks for Plutonium Recycle in LWRs[3-3]

A set of six well-documented experiments with MOX fuel was performed at Battelle Pacific Northwest Laboratories (BNWL) under contract from the Electric Power Research Institute (EPRI) in the mid 1970s. These experiments use fuel that is 2 wt.% Pu, which bounds the plutonium content of SNF (typically about 1 wt.% Pu for well-irradiated fuel). The MOX fuel was made using natural uranium; thus there is a significant amount of U-235 (0.71 wt. %) which may allow for some resonance overlap effects between Pu and U. The experiments were run borated and unborated to allow observation of any large error in the boron cross sections under a MOX environment. Finally, the experiments were done at three different pin pitches, which allow observation of trends with regard to energy spectrum.

Reference 3-4 provides SCALE input decks for these critical experiments.

Table 3-1. MOX Critical Benchmark Experiments for Burnup Credit Method Validation

Experiment Case	Enrichment U-235 (wt. %)	Description	Reference	Lattice Water/Fuel Volume
EPRI UO_2/PuO_2 2wt%PuO_2 7.8wt%Pu-240:				
Experiment 1	0.71	0.700-in. pitch, 0 ppmb	Ref. 3-3	1.20
Experiment 2	0.71	0.700-in. pitch, 688 ppmb	Ref. 3-3	1.20
Experiment 3	0.71	0.870-in. pitch, 0 ppmb	Ref. 3-3	2.53
Experiment 4	0.71	0.870-in. pitch, 1090 ppmb	Ref. 3-3	2.53
Experiment 5	0.71	0.990-in. pitch, 0 ppmb	Ref. 3-3	3.64
Experiment 6	0.71	0.990-in. pitch, 767 ppmb	Ref. 3-3	3.64
SAXTON UO_2/PuO_2 6.6wt%PuO_2 8.6wt%Pu-240:				
Experiment 7	0.71	0.52-in. pitch	Ref. 3-5	1.68
Experiment 8	0.71	0.56-in. pitch	Ref. 3-5	2.16
Experiment 9	0.71	0.56-in. pitch, 337 ppmb	Ref. 3-5	2.16
Experiment 10	0.71	0.735-in. pitch	Ref. 3-5	4.70
Experiment 11	0.71	0.792-in. pitch	Ref. 3-5	5.67
Experiment 12	0.71	1.04-in. pitch	Ref. 3-5	10.75
PNL4976 MOX and UO_2 2wt%PuO_2 7.9wt%Pu-240:				
Experiment 13	MOX 0.71 UO_2 4.31	MOX and UO_2 rods in uniform pattern	Ref.3-6	0.49
PUP UO_2/PuO_2 triangular 2wt%PuO_2 8wt%Pu-240:				
Experiment 14	0.71	0.80-in lattice spacing	Ref. 3-7	1.211
Experiment 15	0.71	0.93-in lattice spacing	Ref. 3-7	1.987
Experiment 16	0.71	1.05-in lattice spacing	Ref. 3-7	2.808
Experiment 17	0.71	1.143-in lattice spacing	Ref. 3-7	3.513
Experiment 18	0.71	1.32-in lattice spacing	Ref. 3-7	5.019
Experiment 19	0.71	1.386-in lattice spacing	Ref. 3-7	5.635
PUP UO_2/PuO_2 triangular 2wt%PuO_2 16wt%Pu-240:				
Experiment 20	0.71	0.93-in lattice spacing	Ref. 3-7	1.987
Experiment 21	0.71	1.05-in lattice spacing	Ref. 3-7	2.808
Experiment 22	0.71	1.143-in lattice spacing	Ref. 3-7	3.513
Experiment 23	0.71	1.32-in lattice spacing	Ref. 3-7	5.019

Experiment Case	Enrichment U-235 (wt. %)	Description	Reference	Lattice Water/Fuel Volume
PUP UO_2/PuO_2 triangular 2wt% PuO_2 24wt% Pu-240:				
Experiment 24	0.71	0.80-in lattice spacing	Ref. 3-7	1.211
Experiment 25	0.71	0.93-in lattice spacing	Ref. 3-7	1.987
Experiment 26	0.71	1.05-in lattice spacing	Ref. 3-7	2.808
Experiment 27	0.71	1.143-in lattice spacing	Ref. 3-7	3.513
Experiment 28	0.71	1.32-in lattice spacing	Ref. 3-7	5.019
Experiment 29	0.71	1.386-in lattice spacing	Ref. 3-7	5.635
PUP UO_2/PuO_2 triangular 4wt% PuO_2 18wt% Pu-240:				
Experiment 30	0.71	0.85-in lattice spacing	Ref. 3-7	1.500
Experiment 31	0.71	0.93-in lattice spacing	Ref. 3-7	1.993
Experiment 32	0.71	1.05-in lattice spacing	Ref. 3-7	2.815
Experiment 33	0.71	1.143-in lattice spacing	Ref. 3-7	3.521
Experiment 34	0.71	1.386-in lattice spacing	Ref. 3-7	5.647
Experiment 35	0.71	1.60-in lattice spacing	Ref. 3-7	7.859
Experiment 36	0.71	1.70-in lattice spacing	Ref. 3-7	9.000
TCA UO_2/PuO_2 3.01 wt.% PuO_2 22.02 wt.% Pu-240				
Experiment 37	0.71	0.72-in lattice spacing	Ref. 3-9	2.42
Experiment 38	0.71	0.72-in lattice spacing	Ref. 3-9	2.42
Experiment 39	0.71	0.72-in lattice spacing	Ref. 3-9	2.42
Experiment 40	0.71	0.77-in lattice spacing	Ref. 3-9	2.98
Experiment 41	0.71	0.77-in lattice spacing	Ref. 3-9	2.98
Experiment 42	0.71	0.77-in lattice spacing	Ref. 3-9	2.98
Experiment 43	0.71	0.88-in lattice spacing	Ref. 3-9	4.24
Experiment 44	0.71	0.88-in lattice spacing	Ref. 3-9	4.24
Experiment 45	0.71	0.88-in lattice spacing	Ref. 3-9	4.24
Experiment 46	0.71	0.97-in lattice spacing	Ref. 3-9	5.55
Experiment 47	0.71	0.97-in lattice spacing	Ref. 3-9	5.55

3.1.2 Saxton MOX Critical Experiments[3-5]

In 1965, Westinghouse conducted critical experiments with some of the MOX pins that were to be placed in the Saxton core. The single region MOX experiments used 6.6 wt.% Pu fuel. Again, natural uranium was used to mix with the plutonium. This set of experiments is still low enriched fuel but allows observation of whether there is a trend in prediction capability with increased plutonium content. The six experiments cover a range of lattice pitches (and hence spectrum) and also contain one experiment which was borated.

Reference 3-4 provides SCALE input decks for these critical experiments.

3.1.3 PNL Mixed Fuel Pin (MOX, UO_2) Critical Experiment[3-6]

BNFL uses Gd in their dissolver at their reprocessing facility. In order to validate its worth, BNFL contracted the Pacific Northwest Laboratory (PNL) to perform critical experiments using dissolved Gd. One experiment was selected that contained both UO_2 and MOX fuel pins. This experiment provides a harder spectrum than most of the MOX experiments. It is fueled with 1174 UO_2 pins with an enrichment of 4.31 wt.% U-235 and 583 MOX pins with 2 wt.% Pu. This experiment provides a good test of cross section treatment. It covers resonance overlap between U-235 and Pu isotopes represented in underburnt fuel.

Reference 3-4 provides SCALE input decks for these critical experiments.

3.1.4 Plutonium Utilization Program (PUP) Critical Experiments[3-7]

In the late 1960's, a series of MOX critical experiments was performed at BNWL. These experiments were performed with various lattice pitches to look for spectral effects. They were also performed with various plutonium vectors so one could determine problems with Pu-240. Twenty-three critical experiments were selected to allow a full analysis of these effects. The critical experiments described in the previous sections all had Pu-240 content of less than 9% of the Pu. This set of experiments contained Pu-240 contents of 8%, 16%, 18%, and 24%.

Reference 3-8 provides SCALE input decks for these critical experiments.

3.1.5 Tank-Type Critical Assembly (TCA) MOX Critical Experiments[3-9]

Between 1972 and 1975 the Tokai Research Establishment of JAERI performed critical experiments with MOX fuel. These experiments are included for two features. First, the plutonium vector is more comparable to SNF than the critical experiments presented in the previous sections. The Pu-240 content is 22%; however, the key difference from the PUP experiments is the 2.04% Pu-242 content, which is much closer to that in spent fuel than the Pu-242 content found in the PUP experiments. The second feature emanates from repeating the experiments after a significant time delay. This delay allowed for Pu-241 decay and the creation of Am-241. Since the experiments are identical, except for the Pu-241 decay (and, of course, the height of water adjusted to maintain criticality), the experiments provide an excellent test of a code system to predict the change in reactivity with time due to Pu-241 decay. Eleven experiments have been documented and accepted as benchmarks by the International Criticality Benchmarking Committee of OECD/NEA. These cases cover four pin pitches and three separate time periods for Pu-241 decay. The plutonium content in each pin is 3 wt.%.

3.2 ESTABLISHMENT OF BIAS AND UNCERTAINTY

Section 5.2 of NUREG/CR-5661 describes the requirements for establishment of bias and uncertainty for the fresh fuel assumption. The same requirements are necessary the burnup credit applications. As stated in the previous section, there are sufficient UO_2 critical experiments for this process. However, there are a limited number of MOX critical experiments. These critical

experiments are simple lattices and do not attempt to model cask dependent features other than spectrum covered by variation in fuel pin pitch.

The criticality characteristics of SNF are bounded by the UO_2 and MOX critical experiments. In order to assure that the bias and uncertainty are conservatively handled, the two sets of experiments are not to be combined, but rather biases and uncertainties are to be determined from each set of critical experiments (UO_2 and MOX) and the more conservative bias and uncertainty are to be used. Since the bias and uncertainty may trend with a parameter the more conservative bias may change from the UO_2 set to the MOX set (or the other way around) as a function of this trending parameter (eg., a spectral index).

As already pointed out, the MOX critical experiment set provides for a limited search for trends. However, statistically significant trends on cask features other than spectrum are rarely observed with calculations of the UO_2 critical experiments. Thus, the limited set of MOX critical experiments is sufficient to find the expected trends. It is important to remember that the UO_2 critical experiments are strongly representative of SNF in that they are still low enriched (mainly U-238) water moderated systems. It would be expected that if a reactivity problem due to some non-actinide isotope used in the cask design existed, it would be seen in the UO_2 critical experiments, and the magnitude of the effect in SNF would be similar. In the unlikely event that a statistically significant bias due to some cask feature (other than spectrum) is observed in the UO_2 critical experiments, the bias for the MOX critical experiments should be conservatively increased to account for this effect.

Since it is not proven that the bias increment would be the same for the MOX critical experiments as it was for the UO_2 critical experiments, some margin from the fission product margin is reserved for this concern. To determine the magnitude of this margin, a review was conducted of analyses of many critical experiments done using many codes and cross section libraries and performed by many organizations. From this review, it was concluded that deviations from the mean for critical experiments are almost never greater than 2% in k_{eff}. Therefore, reserving 2% of the fission product margin for criticality validation should be sufficient to resolve almost any concern. The following observations support the selection of 2% to cover validation concerns:

1) NUREG/CR-6361 contains analysis of 167 LWR-type critical experiments with SCALE 4.3 and the 44-group ENDF/B-V cross-section library.[3-2] The maximum deviation from the mean was 1.35% in k_{eff}.

2) NUREG/CR-6102[3-10] reported 33 experiments analyzed with SCALE 4.2 and the 27GROUPNDF4 library. The maximum deviation from the mean was less than 1.6% in k_{eff}.

3) A set of 34 criticals, previously proposed for burnup credit, have been analyzed with two computer codes and based on three different versions of the ENDF/B libraries. The results of the analyses are detailed in Table 3-2. The maximum deviation between any of these criticals analyzed and any of the reported means is less than 1.6% in k_{eff}.

Table 3-2. Code Comparisons for Thirty-Four Critical Experiments

Experiment[3-4]	SCALE 27G[3-8]	SCALE 44G[3-10]	MCNP ENDF5[3-11]	MCNP ENDF6[3-11]
1	0.9908	0.9947	0.9994	0.9916
2	0.9930	0.9974	0.9981	0.9899
3	0.9925	0.9951	1.0005	0.9904
4	0.9936	1.0007	1.0001	0.9917
5	0.9978	1.0069	0.9970	0.9961
6	0.9980	1.0036	1.0000	0.9951
7	0.9979	0.9936	0.9981	0.9950
8	0.9947	0.9975	1.0073	1.0050
9	0.9987	1.0037	1.0073	1.0065
10	0.9950	0.9994	1.0069	1.0030
11	0.9881	0.9945	1.0018	0.9976
12	0.9943	0.9955	1.0054	0.9987
13	0.9933	0.9967	1.0023	0.9986
14	0.9909	0.9963	0.9969	0.9908
15	0.9960	0.9998	1.0007	0.9966
16	0.9930	1.0008	1.0005	0.9996
17	0.9931	0.9932	0.9980	0.9989
18	0.9955	0.9995	0.9988	1.0002
19	0.9898	0.9942	0.9943	0.9920
20	0.9905	0.9999	1.0020	0.9983
21	0.9923	0.9958	0.9987	0.9980
22	0.9960	0.9991	0.9968	0.9917
23	0.9977	1.0007	1.0026	0.9922
24	1.0032	1.0046	1.0036	0.9959
25	1.0050	1.0079	1.0058	0.9995
26	1.0048	1.0059	1.0087	1.0000
27	1.0073	1.0111	1.0105	0.9992
28	1.0025	0.9985	1.0042	0.9925
29	1.0035	0.9993	1.0054	0.9938
30	0.9998	0.9986	1.0002	0.9960
31	1.0046	1.0034	1.0060	0.9982
32	1.0063	1.0017	1.0063	0.9979
33	1.0076	1.0042	1.0088	1.0001
34	0.9864	1.0021	0.9913	0.9894
Mean	0.9969	0.9999	1.0019	0.9965
Std. Div.	0.0058	0.0044	0.0045	0.0043

4) The Organization for Economic Co-Operation and Development's (OECD) Nuclear Energy Agency's (NEA) Nuclear Science Committee has been active in criticality benchmark analysis and burnup credit benchmarks. Table 3-3 describes three phases of burnup credit benchmarking analyses performed by the OECD/NEA. The calculations were performed by 20 participants from 11 countries using many different computer codes and libraries. In addition, different enrichments, burnups, cooling times, isotope mixes, and burnup profiles were used. Tables 3-4 through 3-6 show the results of these analyses. The maximum 2σ deviation from the mean in these calculations was 1.8 % in k_{eff}.

Table 3-3. Summary of Benchmark Problems Addressed by OECD/NEA Criticality Safety Benchmark Group

Benchmark	Primary Objective	Reference
Phase I-A	Examine effects of 7 major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt. % U-235 at 0, 30, 40 Gwd/MTU and at 1 and 5 year cooling times	3-12
Phase II-A	Examine effect of axially distributed burnup in an array of PWR pins as a function of initial enrichment, burnup and cooling time. Effects of fission products independently examined. Isotopic compositions specified.	3-13
Phase II-B	Repeats study of Phase II-A in a 3-D geometry representative of a conceptual burnup credit transportation container. Isotopic compositions specified	3-14

Table 3-4. Summary of Phase I-A Results: Average Multiplication Factor (k_{eff})

Case	Initial Enrichment	Burnup GWd/MTU	Cooling Time (yr)	Actinides	Fission Products	k_{eff} (2σ)	2σ (%)
1	3.6 wt %	Fresh	N/A	All	All	1.4378(0.0175)	1.2
2	3.6 wt %	30	1	All	All	1.1080(0.0194)	1.8
3	3.6 wt %	30	5	All	All	1.0758(0.0185)	1.7
4	3.6 wt %	30	1	All	No	1.2456(0.0107)	0.86
5	3.6 wt %	40	1	All	No	1.1885(0.0110)	0.93
6	3.6 wt %	30	5	All	No	1.2284(0.0109)	0.89
7	3.6 wt %	40	5	All	No	1.1657(0.0099)	0.85
8	3.6 wt %	30	1	Major	No	1.2635(0.0108)	0.85
9	3.6 wt %	30	5	Major	No	1.2566(0.0109)	0.87
10	3.6 wt %	30	1	All	Major	1.1402(0.0169)	1.5
11	3.6 wt %	30	5	All	Major	1.0638(0.0170)	1.6
12	3.6 wt %	40	1	All	Major	1.1123(0.0164)	1.5
13	3.6 wt %	40	5	All	Major	1.0240(0.0156)	1.5

Table 3-5. Summary of Phase II-A Results: Average Multiplication Factor (k_{eff})

Case	Initial Enrichment	Burnup GWd/MTU	Cooling Time (yr)	Fission Products	Burnup Profile	k_{eff} (2σ)	2σ (%)
1	3.6 wt %	Fresh	N/A	N/A	N/A	1.4335 (0.0217)	1.5
2	3.6 wt %	10	1	Yes	Yes	1.3053 (0.0161)	1.2
3	3.6 wt %	10	1	Yes	No	1.3126 (0.0159)	1.2
4	3.6 wt %	10	1	No	Yes	1.3607 (0.0175)	1.3
5	3.6 wt %	10	1	No	No	1.3665 (0.0174)	1.3
6	3.6 wt %	30	1	Yes	Yes	1.1360 (0.0155)	1.4
7	3.6 wt %	30	1	Yes	No	1.1358 (0.0138)	1.2
8	3.6 wt %	30	1	No	Yes	1.2339 (0.0129)	1.0
9	3.6 wt %	30	1	No	No	1.2419 (0.0119)	0.96
10	3.6 wt %	30	5	Yes	Yes	1.1160 (0.0144)	1.3
11	3.6 wt %	30	5	Yes	No	1.1062 (0.0136)	1.3
12	3.6 wt %	30	5	No	Yes	1.2176 (0.0119)	0.98
13	3.6 wt %	30	5	No	No	1.2256 (0.0113)	0.9
14	4.5 wt %	Fresh	N/A	N/A	N/A	1.4783 (0.0232)	1.6
15	4.5 wt %	30	1	Yes	Yes	1.1996 (0.0151)	1.3
16	4.5 wt %	30	1	Yes	No	1.2025 (0.0161)	1.3
17	4.5 wt %	30	1	No	Yes	1.2972 (0.0145)	1.1
18	4.5 wt %	30	1	No	No	1.3064 (0.0139)	1.1
19	4.5 wt %	50	1	Yes	Yes	1.0838 (0.0175)	1.6
20	4.5 wt %	50	1	Yes	No	1.0584 (0.0136)	1.3
21	4.5 wt %	50	1	No	Yes	1.1999 (0.0121)	1.0
22	4.5 wt %	50	1	No	No	1.1983 (0.0116)	0.97
23	4.5 wt %	50	5	Yes	Yes	1.0543 (0.0156)	1.5
24	4.5 wt %	50	5	Yes	No	1.0123 (0.0135)	1.3
25	4.5 wt %	50	5	No	Yes	1.1800 (0.0104)	0.88
26	4.5 wt %	50	5	No	No	1.1734 (0.0096)	0.82

Table 3-6. Summary of Phase II-B Results: Average Multiplication Factor (k_{eff})

Case	Initial Enrichment	Burnup GWd/MTU	Cooling Time (yr)	Fission Products	Burnup Profile	k_{eff}	2σ (%)
1	4.5 wt %	Fresh	N/A	N/A	N/A	1.1257	1.3
2	4.5 wt %	30	5	Yes	No	0.8934	0.7
3	4.5 wt %	30	5	No	No	0.9716	1.0
4	4.5 wt %	30	5	Yes	Yes	0.8953	1.0
5	4.5 wt %	30	5	No	Yes	0.9647	1.1
6	4.5 wt %	30	5	Yes	No	0.7641	0.5
7	4.5 wt %	30	5	No	No	0.8737	0.7
8	4.5 wt %	30	5	Yes	Yes	0.7933	0.8
9	4.5 wt %	30	5	No	Yes	0.8791	1.0

Due to these observations, 2% in k_{eff} margin should easily provide sufficient conservatism to compensate for undetected biases, trends, or uncertainties associated with criticality validation. (The final rack of fission product margin is shown in Chapter 7.)

3.3 ESTABLISHMENT OF RANGE OF APPLICABILITY

The range of applicability for burnup credit should be handled in the same manner as established in NUREG/CR-5661 for the fresh fuel assumption. The range should be demonstrated primarily based on the UO_2 experiments, just as in the fresh fuel assumption. The $_2$UO (having no plutonium) and MOX (having 2 wt.% Pu or greater) critical experiments bound SNF in gross plutonium content. Individual isotopes are also generally bounded as illustrated in Figures 3-1 and 3-2. The range of applicability for the MOX experiments based on a selected spectral index can be established. Due to the lack of variation in the MOX critical experiments, all other range of applicability issues must be based on the UO_2 critical experiments and the 2% in k_{eff} fission product margin.

3.4 ESTABLISHMENT OF ACCEPTANCE CRITERIA

There is nothing unique about burnup credit in the establishment of acceptance criteria. NUREG/CR-5661 adequately covers the topic and refers to NUREG/CR-6361 for details on determining the upper subcriticality limit (USL). NUREG/CR-6361's "USL Method 1" is fully endorsed by this topical report for use in burnup credit. The only remaining issue is whether the 5% margin of subcriticality specified in NUREG/CR-5661 is appropriate. Since this topical report generally uses bounding analytical techniques, the margin to criticality is greater than 5%. The "fresh fuel assumption" also normally adds margin above 5%, but this margin is not assured since there are no burnup requirements. In a burnup credit cask, the additional margin is assured by a burnup requirement and a verification measurement for that burnup. It has been decided that the traditional 5% margin to subcriticality will be applied for actinide-only burnup credit.

3.5 DEMONSTRATION OF USE OF THE MOST LIMITING USL

The criticality validation requires the cask vendor to establish a set of UO_2 critical experiments appropriate for its cask and fuel to be loaded. In addition, the cask vendor will analyze the 47 MOX critical experiments. The cask vendor will then determine if any trends exist on the appropriate parameters with the UO_2 data and will look for trends with regard to a spectral index in the analysis of the MOX critical experiments. The cask vendor will establish whether the cask and fuel intended to be loaded are within the range of applicability of the criticality validation. Finally, the cask vendor will determine the USL from both the UO_2 and MOX critical experiment analysis, and use the most limiting of the two.

For illustration of the use of the most limiting USL, an analysis was performed for a set of UO_2 and MOX critical experiments using SCALE 4.2 and the 27BURNUPLIB.[3-15] (Although the values presented are based on an actual analysis, they do not correspond to an appropriate set of UO_2 critical experiments for any particular cask and do not represent the calculation of the full set of MOX critical experiments.) For this analysis, the average lethargy for absorption was selected as the spectral trending parameter. Statistical trending analyses were performed on each experimental subset (UO_2 and MOX experiments) individually. Only one statistically significant trend was observed. This was a trend against the average lethargy for absorption for the MOX

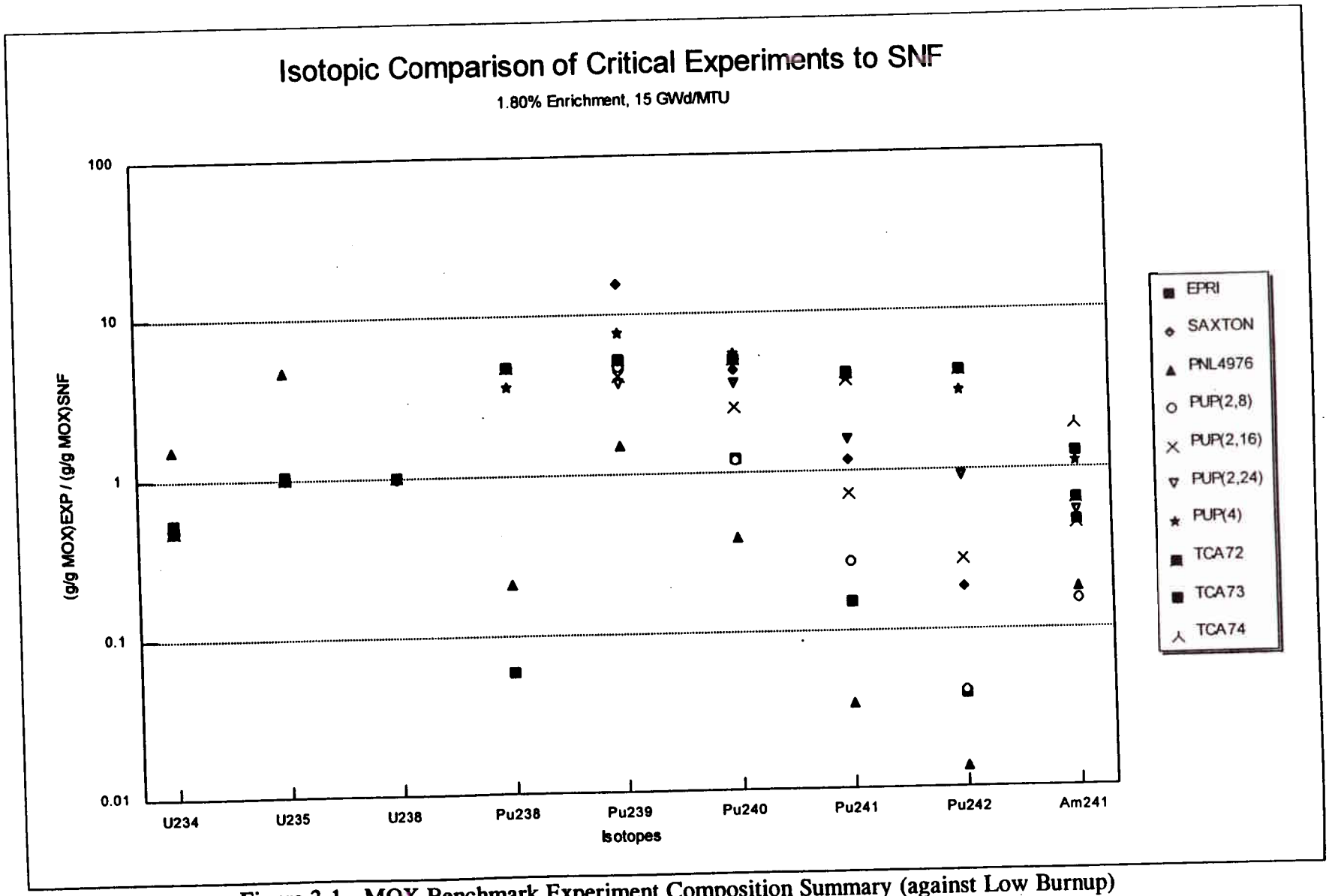


Figure 3-1. MOX Benchmark Experiment Composition Summary (against Low Burnup)

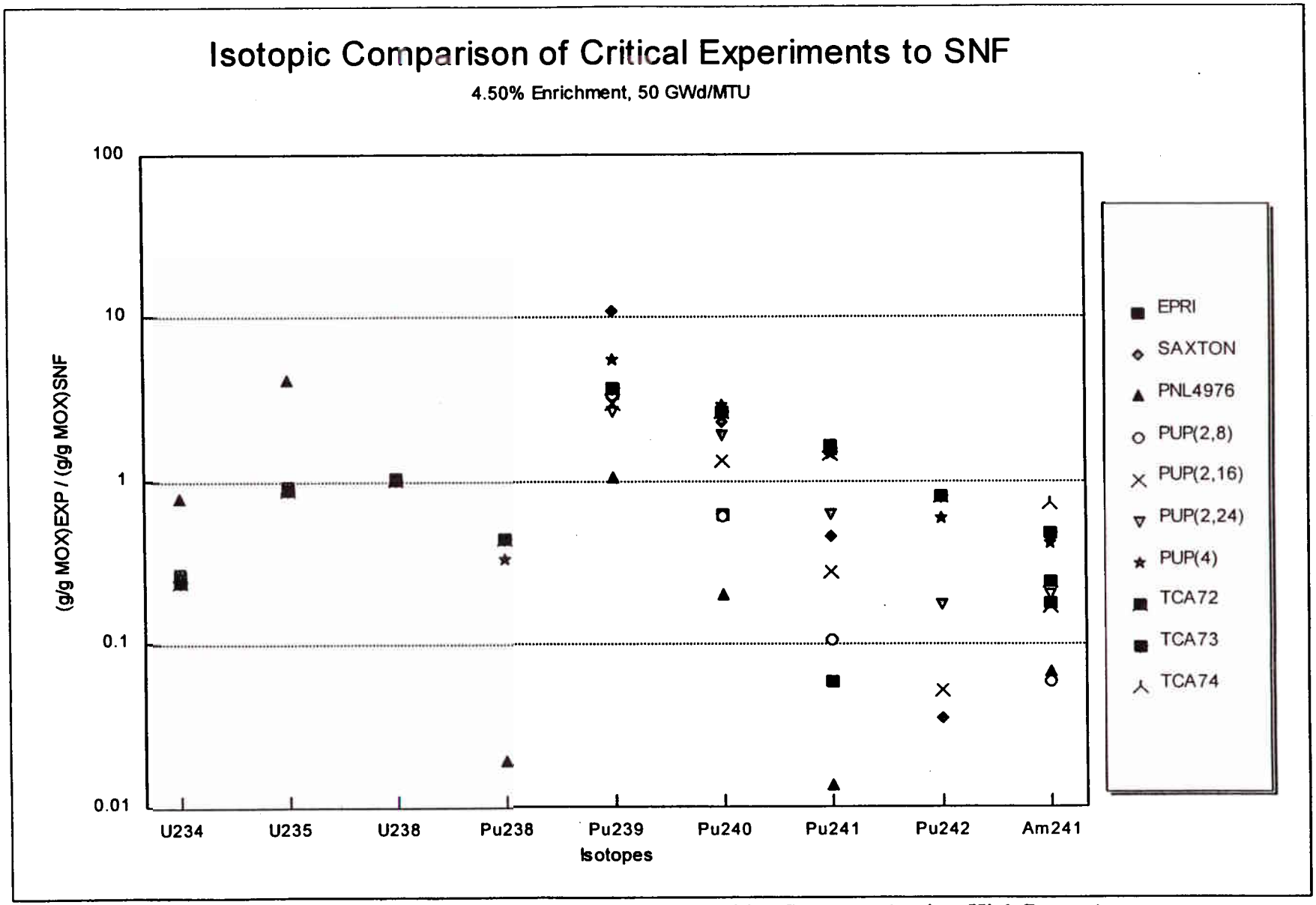


Figure 3-2. MOX Benchmark Experiment Composition Summary (against High Burnup)

subset. The benchmark calculation k_{eff} results and USL calculation parameters are plotted in Figure 3-3 for the UO_2 subset and Figure 3-4 for the MOX subset.

Although plotted against the ALA in Figure 3-4, the UO_2 subset did not exhibit a statistically significant trend against any of the required trending parameters. The uncertainty for this subset was 0.0056 Δk, much lower than the administrative margin of 0.05 Δk_m, affirming that 0.05 is adequate. For the MOX subset, the uncertainty observed was lower than 0.01 Δk for the ALA range of interest. Figure 3-5 presents the final, combined USL.

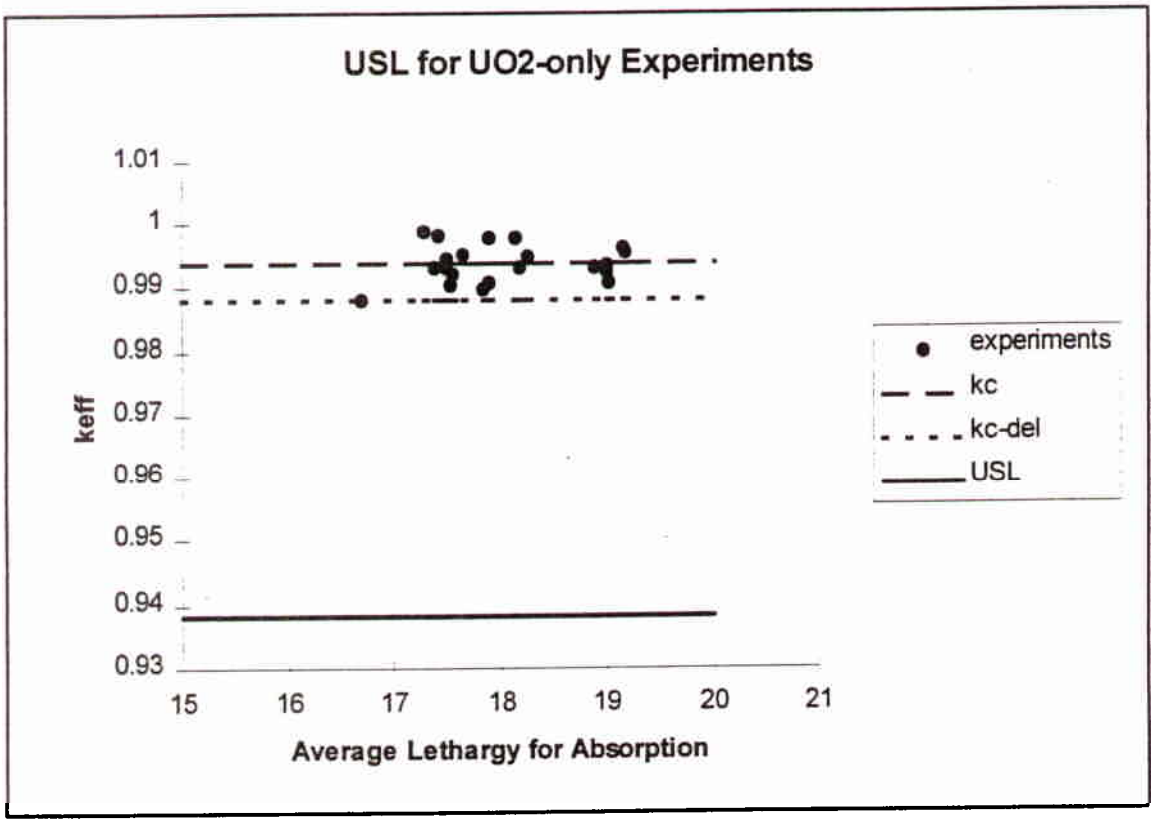


Figure 3-3. Upper Safety Limit (USL) for UO_2 Experiment Subset

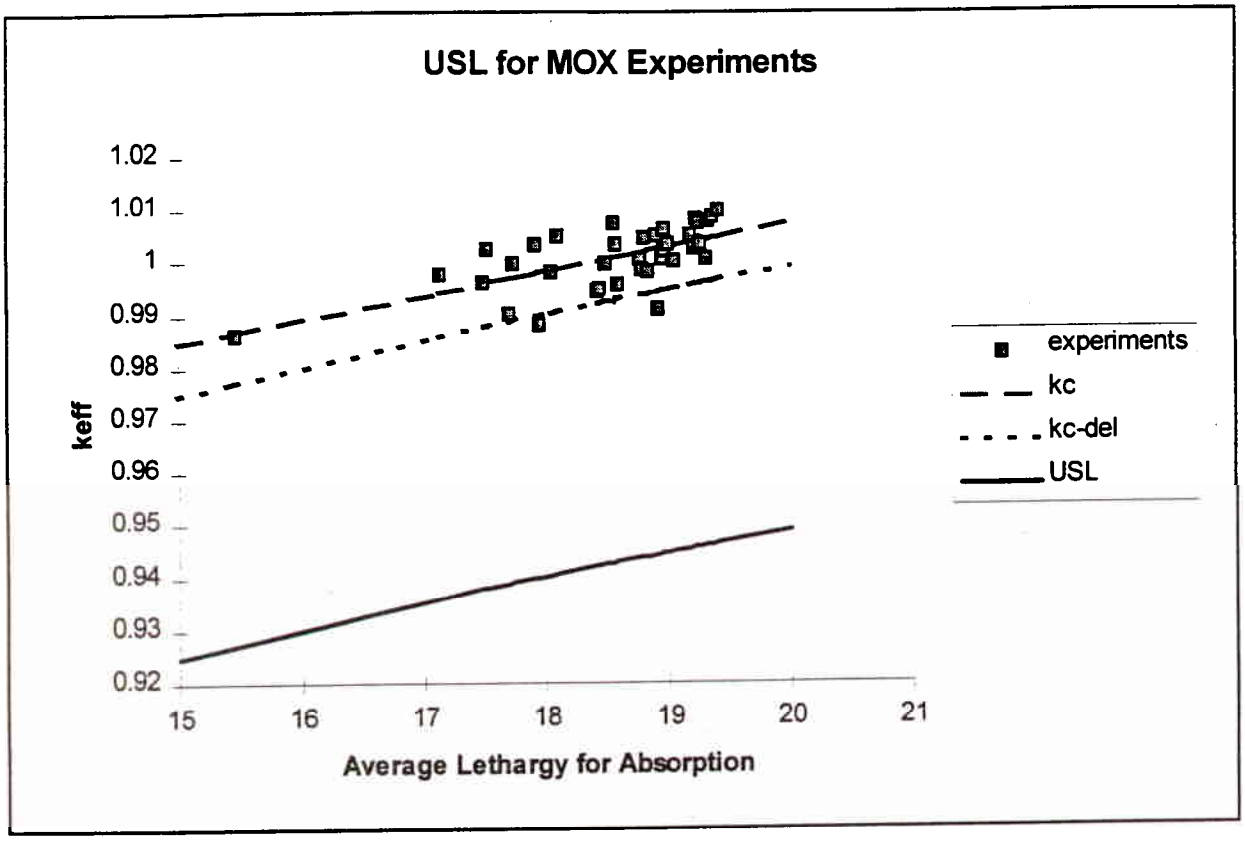


Figure 3-4. Upper Safety Limit (USL) for MOX Experiment Subset

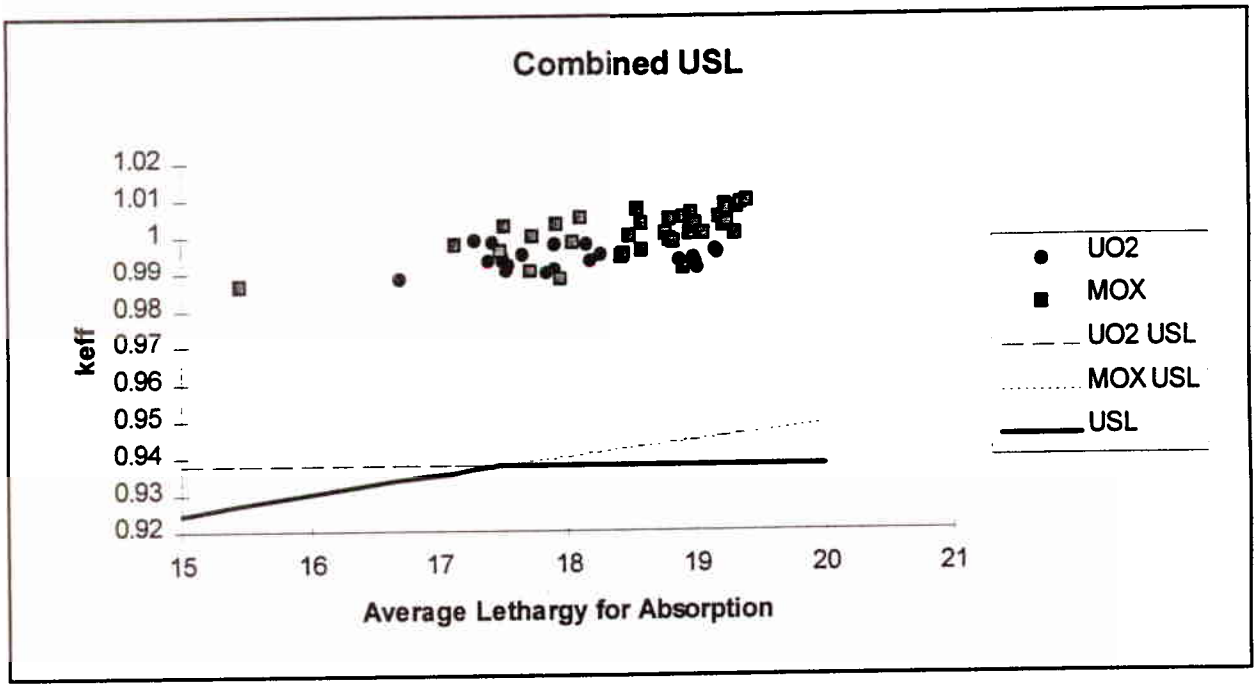


Figure 3-5. Final Upper Safety Limit (USL)

3.6 SUMMARY AND CONCLUSION

This chapter generally affirms the positions taken in NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages." Some modifications are required for burnup credit. Actinide-only isotopes in SNF are bounded by UO_2 and MOX critical experiments. The upper subcriticality limit, USL, for actinide-only burnup credit is established as the most limiting of the USL from the UO_2 and MOX analysis. All of the analysis for the UO_2 set should follow the procedures established for the fresh fuel assumption in NUREG/CR-5661. Since there are a more limited number of MOX critical experiments, they have been selected generically to cover all casks and PWR fuels. The 47 MOX critical experiments cover a range of isotopic distributions of plutonium and americium. They also cover a range of spectrum. Since there is insufficient data to test all possible trends with the MOX critical experiments, 2% in k_{eff} of the fission product margin is reserved for criticality validation concerns. This 2% is generously conservative since the maximum variation from the mean k_{eff} for calculation of relevant critical experiments from numerous national and international benchmarks rarely reaches 2%.

This topical report specifically seeks NRC acceptance of: 1) the selection of the 47 MOX critical experiments for actinide-only burnup credit analysis, and 2) the use of the most limiting USL from the UO_2 and MOX criticality experiment analysis. This topical report also seeks confirmation that the methods for criticality validation established in NUREG/CR-5661 are applicable for burnup credit with the supplement of the MOX critical experiments as specified above.

4. ANALYSIS AND MODELING PARAMETERS

This chapter provides a method to perform bounding depletion and criticality analyses of spent PWR fuel. Chapters 2 and 3 describe how the depletion and criticality models are validated and how they are to be used to assure conservative results. This chapter provides a method to develop conservative inputs to these validated models. Sensitivity analyses have been performed to determine the input parameters that can have a significant effect on the depletion and criticality analyses.[4-1] For depletion analysis, these input parameters are specific power, dissolved boron, moderator temperature, fuel pellet temperature, and control rod insertion history. (Burnable absorber properties are considered assembly design features and are discussed in Chapter 5.) For the criticality analysis, these input parameters are moderator density, fuel temperature, axial burnup distribution, and horizontal burnup distribution.

4.1 LIMITING PARAMETERS IN THE CALCULATION OF ISOTOPIC CONCENTRATIONS

The parameters discussed in this section are independent of the specific design of an SNF package system and only effect the isotopic concentration of the fuel to be loaded in the package. They are determined by the operating history at the nuclear power plant. The parameters are the specific power level, operating time at that power, the dissolved boron concentration (parts per million boron, ppmb), the water moderator temperature, the fuel pellet temperature, and control rod operation.

4.1.1 Specific Power

The specific power level of the assembly determines the rate of production of heavy elements and fission products of interest to burnup credit. A number of these isotopes undergo significant radioactive decay during the burning of the fuel. In addition, isotopes have a neutron capture cross section so that a quantity of the isotope is transmuted during the burning process. The rate of production compared to the rate of decay and transmutation determines the isotopic concentration in the spent nuclear fuel.

An increase in specific power results in two changes: (1) increase in neutron flux used for fuel depletion and (2) decrease in fuel depletion time (to achieve a same burnup). The decrease in fuel depletion time has a negligible effect on the majority of the actinides because of their long half-lives. However, Pu-241 is affected because of its short half-life of 14.4 years. Essentially, Pu-241 has less time to β-decay to Am-241. Therefore, the concentration of Pu-241 increases as the specific power increases. Consequently, the concentration of Am-241 decreases as the specific power increases because the main production chain of Am-241 is the β-decay of Pu-241. In addition, the concentration of Pu-238 decreases as the specific power increases. Increase in neutron flux indirectly effects the actinide concentration. The equilibrium concentration of Xe-135 increases as the neutron flux increases. This increase in Xe-135 concentration hardens the neutron spectrum[4-1] to which a fuel assembly is exposed. The spectrum hardening causes increased absorption in U-238 by resonance capture and consequently increases the concentration of fissile plutonium isotopes. Subsequently, U-235 is depleted less as more fissions occur in

plutonium isotopes. Ultimately, the net effect of these changes is the increase in spent fuel reactivity with respect to specific power.

A comparison[4-1] of the reactivity of PWR fuel versus the specific power level at which the irradiation occurred is shown in Table 4-1. In these comparisons, the burnup is fixed, and the specific power level is varied from 10 to 50 MW/MTU. Reactivity calculations were performed with U-235 initial enrichments of 3.0, 3.6, and 4.5 w/o U-235 and burnups of 10, 30, and 50 GWd/MTU to encompass the typical range of enrichments and burnups for PWR fuel assemblies. Fuel and moderator temperatures were fixed to limit the number of variables being studied. Table 4-1 provides reactivities at a 3.6 w/o U-235 initial enrichment and 30 GWd/MTU burnup. The reactivities for the other enrichments and burnups displayed trends consistent with the 3.6 w/o, 30 GWd/MTU trend and are not shown in Table 4-1 for clarity. Inspection of the table shows that a higher specific power level assumption results in a higher discharge reactivity.

Table 4-1. k_{inf} versus Specific Power

Specific Power (MW/MTU)	k_{inf} (3.6 w/o, 30 GWd/MTU)
10	1.19194
15	1.19503
20	1.19671
25	1.19781
30	1.19855
35	1.19913
40	1.19950
45	1.19982
50	1.20008

The dependence of reactivity upon fuel cycle variations was also investigated.[4-1] Eleven variations of a three-cycle burnup were evaluated ranging from constant power to variable specific power levels. The cycle variations are illustrated in Figure 4-1, and the resulting reactivities are tabulated in Table 4-2. The reactivity after a continuous burning (Case 1, No Downtime, which is equivalent to a single cycle) is higher than other cases because the omission of the time between cycles (used to reload fuel in the reactor and perform maintenance) effectively shortens the cooling time and contributes to the increased reactivity. Furthermore, the power in an assembly depends upon its position within the reactor core, which is typically changed each cycle, so that cycles with higher and lower power were evaluated (Cases 9 through 11). A higher power (120% of the

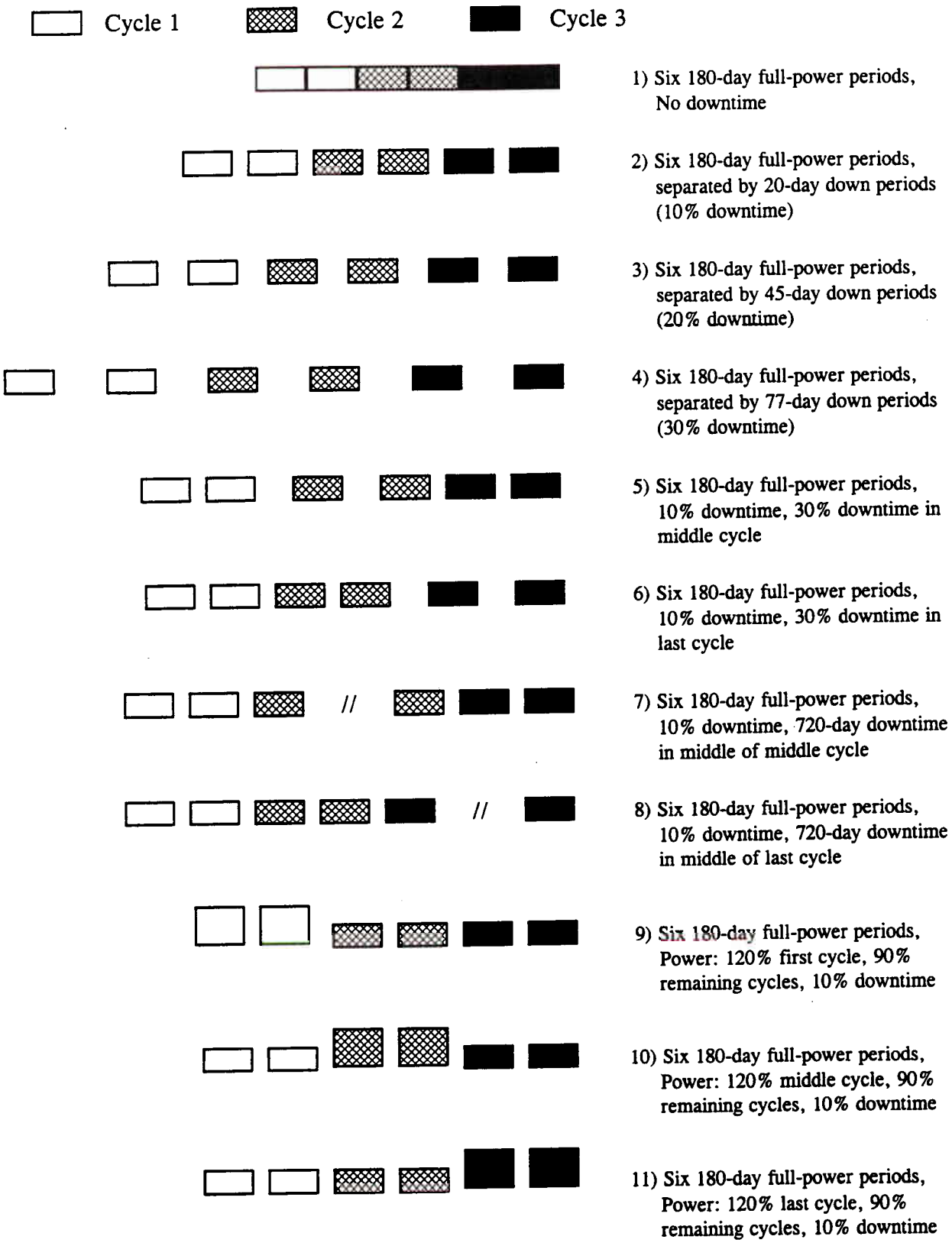


Figure 4-1. Cycle Operating History Cases

average in the core) in the third cycle (Case 11) results in a higher reactivity for the discharged fuel, again due to an effective shortening of the cooling time for most of the radionuclides produced during the burning. The number of possible variations of cycle power and cycle lengths is large; therefore, a single irradiation cycle (Case 1) combined with higher than average power (Case 11) is required to provide a conservative cycle model. Specific powers for PWR fuel are typically 45 MW/MTU or less[4-2], and applying a 120% factor yields a specific power for the single cycle model of 54 MW/MTU. Thus, to ensure conservative results, fuel depletion analyses shall be performed with a single cycle at 60 MW/MTU.

Table 4-2. k_{inf} versus Cycle Operating History

Case	k_{inf} (3.0 w/o, 30 GWd/MTU)
1	1.14391
2	1.14370
3	1.14333
4	1.14287
5	1.14355
6	1.14308
7	1.14237
8	1.13890
9	1.14312
10	1.14360
11	1.14448

4.1.2 Dissolved Boron Effects

Boron is dissolved in the reactor coolant of PWRs so that the reactivity change due to burnup, and excess initial reactivity, can be adjusted without the use of control rods. This provides significant core design benefits. At the beginning of an irradiation cycle, the boron concentration is at a maximum level. As the fuel is burned and the core becomes less reactive, the boron concentration is reduced. A higher concentration of boron causes a harder spectrum in the reactor core and fuel assembly, and the lower thermal flux component reduces the U-235 use. Therefore, when the fuel assembly is discharged from the reactor, it retains a greater portion of the initial U-235. Enhanced plutonium utilization includes greater production of Pu-239 by U-238 neutron capture because the plutonium value of υ (number of neutrons produced per fission) is greater. In

addition, the total recoverable energy per fission is approximately 4% greater for Pu-239 than for U-235; consequently, less total fissioning is required to maintain a given specific power level when Pu-239 is burned. Thus, a discharged PWR fuel assembly contains a higher effective (U-235 and fissile plutonium) enrichment and is more reactive when placed in the SNF package system. Enhanced plutonium production also includes increased production of Pu-240 and Pu-242, which absorb neutrons, so there is some counteracting decrease in reactivity. However, this effect is smaller than the reactivity increase caused by the U-235 and fissile plutonium. Figure 4-2 shows the increase in spent fuel reactivity with respect to boron concentration.[4-2] Therefore, the use of the maximum value that the cycle average ppmb can attain for a given fuel type results in a conservative prediction of the reactivity effects of dissolved boron.

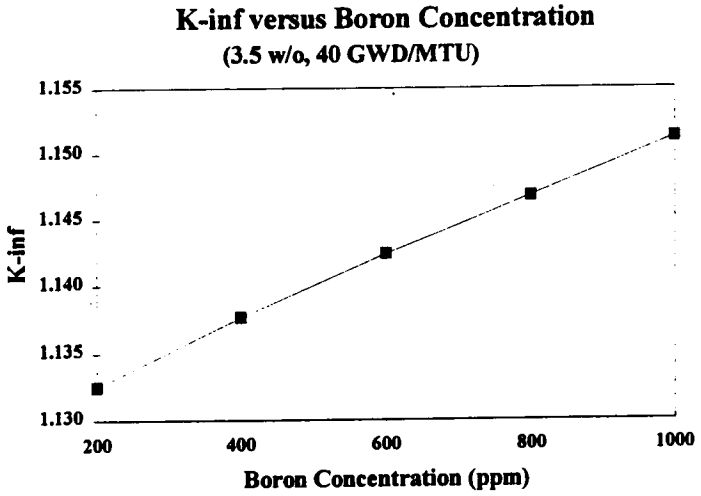


Figure 4-2. k_{inf} versus Boron Concentration

In practice, the boron concentration is adjusted continuously as the fuel is burned, and the critical boron letdown curve is generated as part of normal fuel reload analysis. The average boron concentration is to be found by integrating the boron letdown curve with respect to time and dividing it by the cycle length. The maximum average boron concentration is to be identified from all assemblies for which the loading curve applies. The maximum average boron concentration shall be used for burnup credit analyses.

4.1.3 Moderator Temperature

The neutron spectrum in a reactor core, and the fuel assembly is influenced by the moderator density during reactor operation. For a given reactor pressure, the moderator density decreases as the moderator temperature increases unless boiling occurs. As the moderator density decreases, there is less hydrogen between the fuel rods to slow down neutrons, and a shift toward a harder spectrum is the result. The spectrum hardening increases the resonance capture in U-238. The increase of resonance capture in U-238 results in increased fissile plutonium production.

Consequently, this leads to increased fissions in plutonium and decreases U-235 depletion. The net effect is an increase in spent fuel reactivity[4-2] as shown in Figure 4-3.

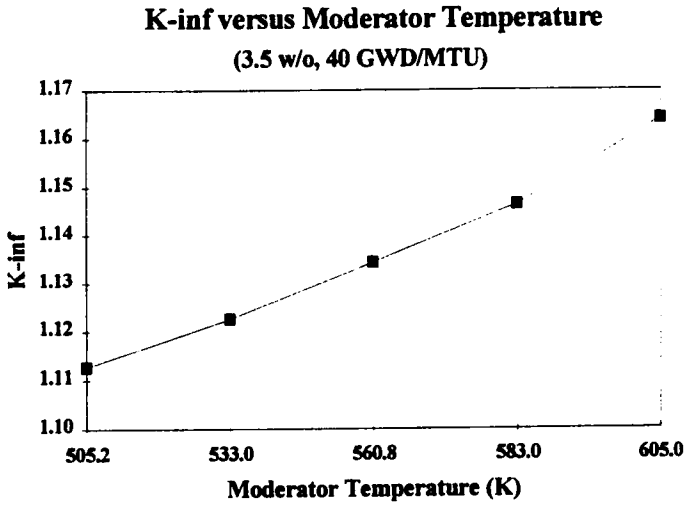


Figure 4-3. k_{inf} versus Moderator Temperature

The moderator temperature increases from the bottom to the top of the core. Thus, the use of average core outlet temperature appears to bound the moderator temperature conservatively. Applying the average core outlet temperature over the entire fuel length and for the entire depletion time provides adequate assurance of bounding treatment.[4-2] The maximum average core outlet temperature and its equivalent density are to be identified from assemblies for which the loading curve applies. The maximum average core outlet temperature shall be used for burnup credit analyses.

4.1.4 Fuel Pellet Temperature

At reactor startup, the fuel pellet temperature rises when the fuel begins to generate the heat that will power the steam turbine to produce electricity. The fuel pellet temperature rise causes the U-238 resonance cross sections to become Doppler broadened, which in turn increases the probability of resonance capture within the pellet. As more U-238 resonance captures occur, more Pu-239 and Pu-241 are produced. This subsequently leads to increased fissions in plutonium and decreases U-235 depletion. Therefore, as shown in Figure 4-4, spent fuel is more reactive after loading in an SNF package when higher pellet temperatures are used in the depletion calculations[4-2].

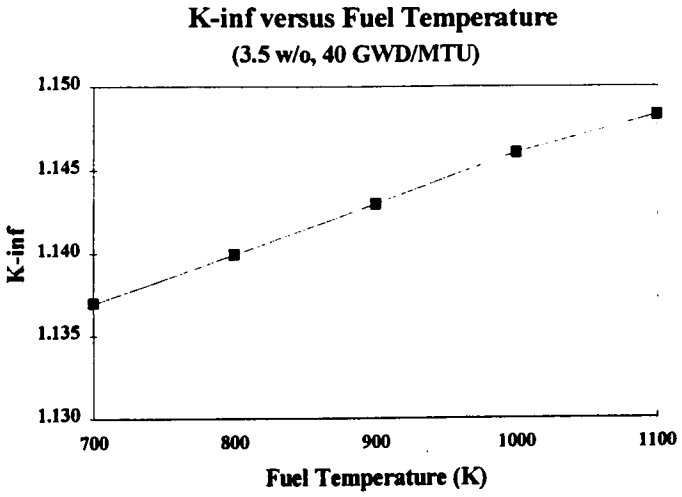


Figure 4-4. k_{inf} versus Fuel Temperature

The nominal average pellet temperature should be calculated based on a reactor rated linear power multiplied by the radial peaking factor limit. A sufficiently conservative value[4-2] can be obtained using a uniform axial power distribution and taking the average pellet temperature from the top of the fuel assembly. For gap conductance and thermal conductivity, the burnup that results in the highest fuel temperature should be used. The maximum average pellet temperature is to be identified from all assemblies for which the loading curve applies. The maximum average pellet temperature shall be used for burnup credit analysis.

4.1.5 Control Rods

Control rods are composed of highly neutron-absorbing materials and are used for rapid shutdown of a nuclear chain reaction. It is possible to use control rods for long-term criticality control (compensation for a reactivity decrease due to burnup) as in BWRs, but PWRs rely on soluble boron to achieve this task. The control rods in PWRs are normally reserved to maintain reactor shutdown margins. For this reason, the control rods in PWRs are normally at their fully-withdrawn positions during fuel depletion. However, some control rods may be inserted to their "bite positions" so that the rate of a negative reactivity introduction is large enough to aid response to power demand.

An exception is partial-length axial power shaping rods used at some plants to control axial power peaking. These control rods are shorter than the active fuel length and are usually positioned at the axial mid-plane The assemblies subjected to axial power shaping rods, however, are shuffled from cycle to cycle based on normal fuel management practices. Thus, it is unlikely to find a single assembly exposed to axial power shaping rods for more than one cycle.

Qualitatively, the presence of control rods leads to a harder neutron spectrum and consequently higher SNF reactivity. Fortunately, control rods are usually fully withdrawn during fuel depletion, and an assembly subjected to fully-inserted axial power shaping rods in one cycle is usually moved to another position in the next cycle.

This topical report does not require cask vendors to model control rods during fuel depletion. Rather, a portion of fission product margin is used to cover any control rod effect. To determine the bounding estimates, k_{eff} is calculated with control rods fully inserted for 15 GWd/MTU. 15 GWd/MTU is greater than most long cycles when considering the flux depression due to the control rods. At low enrichments, the effect is greatest since the plutonium produced by the hardened spectrum is competing with less U-235. At 3.0 w/o enrichment, the reactivity increase is 3.3%. The k_{eff} differences between assemblies with and without control rods for 3.0, 3.6 and 4.5 wt.% U-235 are tallied against available fission product margins in Table 7-4.

This approach to control rod depletion effects is very conservative. For the axial shaping rod cases, the effect would be in the center of the core where the rods would be positioned, and not at the top end of the assembly where the reactivity dominates in spent fuel. In addition, the flux distribution for these assemblies is flatter than that used for cask analysis; therefore, the conservative treatment for these assemblies will compensate for the control rod depletion effect. For assemblies under the lead control bank, the assumption that the control rods are fully inserted is very conservative. Many plants run the control rods all the way out. Others run at the location that produced about $2x10^{-5}$ delta-k per control rod step known as the "bite" position. The bite position is generally no deeper than the top two nodes of a uniform 18 node analysis of the assembly. Although the top of the assembly is the high worth location, this effect covers, at most, the top half of the flux peak. This would suggest that the effect is over estimated for these assemblies by a factor of at least two.

On top of the conservatism in the estimate of the reactivity for any assembly, the approach is conservative since only a small fraction of the assemblies would be influenced by the control rods. The lead control bank is the only control bank allowed to be inserted at significant power operation. In Westinghouse plants, the lead control bank is over only 4, 5, 8 or 9 assemblies. This represents, at most, 7% of the assemblies. In a normal cask loading, the effect of the control rods on depletion would be significantly diluted by assemblies which were not influenced by control rod insertion.

4.1.6 Summary of Limiting Parameters

To provide a consistent set of modeling parameters for actinide-only burnup credit analyses, the depletion isotopic calculations performed for burnup credit analyses shall use a specific power of 60 MW/MTU applied for a single cycle of sufficient length to produce the desired burnup. The boron concentration used shall be the maximum value of the cycle average ppmb appropriate for the assembly type being analyzed, and the moderator temperature shall be the maximum core-average outlet temperature. The fuel pellet temperature shall be the maximum average pellet temperature for the given assembly design. These values shall be appropriate for the PWR assembly type being analyzed and represent the maximum, most conservative values. The values

shall be recorded on the burnup credit loading curve as presented in Chapter 5 of this topical report. Cask vendors are not required to model control rods. Portions of fission product margin are reserved to cover unlikely fuel depletion with inserted control rods.

4.2 SNF PACKAGE DESIGN SPECIFIC EFFECTS

The effects discussed in this section are dependent upon the specific design of a cask system. These modeling parameters include the density of the water moderator in the SNF package, the fuel temperature and the fuel assembly axial, and the horizontal burnup profile.

4.2.1 Moderator Density

Criticality safety analyses must consider optimum moderator density to ensure that the most reactive configuration is evaluated (i.e., a fully flooded cask must be evaluated per 10 CFR §71.55). PWR assemblies are designed to be under-moderated, and reductions in water density from the maximum value of 1.0 g/cc result in a decrease in the k_∞ of the fuel. The maximum reactivity for spent fuel in storage or transport fuel baskets is thus usually achieved at 1.0 g/cc, the maximum density of water. However, for systems in which the water contains dissolved boron and for new fuel storage racks, a reactivity maximum may occur at lower densities. Typical maximum reactivity densities for new fuel storage racks are approximately 0.1 g/cc. Spent fuel baskets in borated water achieve a maximum reactivity at around 0.7 to 0.8 g/cc.

In an SNF package design, the most reactive moderator density varies depending upon the detailed design of the spent fuel basket. Significant differences in sensitivity to moderator density occur between baskets with closely-packed fuel arrays and baskets that include flux traps. The addition of a water gap flux trap to the basket structure could cause a reactivity maximum at a density less than 1.0 g/cc because even though low water density decreases the moderation of neutrons within the fuel, it also decreases the effectiveness of the flux trap. The flux trap works by slowing down fast neutrons within the water gap, causing them to be absorbed by neutron absorbers such as B-10 within the structure of the fuel basket. The low water density decreases the moderation of fast neutrons within the flux trap so that more neutrons pass between adjacent assemblies, increasing the reactivity of the SNF package.

Burnup credit analyses must consider the effects of moderator density from 0 to 1.0 g/cc within the spent fuel package. Given the sensitivity of SNF multiplication factors on the moderator density, the full moderator density range must be considered. Especially in the low moderator density range, a small density increment should be adopted to assure that a peak density is not missed. In addition, the potential for uneven and preferential flooding, which might decrease the effectiveness of criticality control design features, must be addressed in the same manner as in the fresh fuel assumption. The moderator density effects must be evaluated with zero burnup at low enrichments (the maximum fresh fuel enrichment limit for the SNF package) and at high enrichments (the highest enrichment evaluated for the package) with the associated burnup from the burnup credit loading curve. If these evaluations indicate that a reactivity maximum exists at any density but 1.0 g/cc, an optimum moderator density search is required at all enrichments evaluated for the burnup credit loading curve.

4.2.2 Fuel Temperature

When a cask loaded with spent fuel reaches thermal equilibrium, it can be significantly hotter than when first loaded. The increase in fuel temperature increases the resonance capture of neutrons in U-238 and decreases the multiplication factor of the SNF[4-2]. Therefore, an ambient temperature of 20°C (293K) should be used as the fuel temperature in SNF casks regardless of the thermal equilibrium temperatures expected in normal and accident conditions.

4.2.3 Axial Burnup Profile

The axial power peaking effect caused by neutron leakage from the ends of the finite-length fuel assembly produces an axial profile in the burnup. This axial variation in burnup can be accurately described by adopting axial multiple zones of varying burnup within a fuel assembly. However, the fuel assembly modeled with an axially uniform assembly average burnup results in over-prediction of reactivity in the fuel mid-region and under-prediction in the fuel end region. The reactivity difference between the axially burnup-dependent analysis and the uniform analysis is commonly known as the "end effect," and the relative neutron importance of the over-predicted fuel mid-region and the under-predicted fuel end region determines the sign and magnitude of the end effect.

4.2.3.1 Limiting Axial Burnup Profile

An example of the axial profile of spent fuel is illustrated by the measurement of Cs-137 as shown in Figure 4-5.[4-3] The shape of the burnup profile is a flattened cosine, with a peak from 1.1 to 1.2 times the average value of the burnup, and a burnup at the fuel rod ends that equals from 50 to 60% of the average value. Details of the calculational modeling approach used for the end effect are discussed below. The axial profile for each individual spent fuel assembly will vary somewhat from this profile depending on the specific power history of the assembly.

A PWR axial burnup profile database[4-4] has been compiled to study the effect of different axial burnup profiles on the end effects. The database includes 3169 axial burnup profiles from five different PWR fuel types. The profiles are calculated from fuel management codes and represent 20 different PWR reactors and 105 operating cycles. The profiles are tabulated as 18 normalized, equal-size nodes. The end effect of an infinite fuel array has been analyzed using these profiles[4-5] and the results are shown in Figure 4-6. The end effect reactivity illustrated in Figure 4-6 is defined as $(k_{18\ nodes} - k_{uniform})/k_{uniform}$. The bounding axial profile analysis[4-5], however, implicitly included fission products in addition to actinides. The end effect profile rankings determined by the bounding profile analysis[4-5] have been repeated on selected profiles and confirmed in a separate analysis[4-2] using the actinide-only methodology. In addition, the limiting profiles to be used with actinide-only burnup credit methodology have been determined in the same study.[4-2] Table 4-3 shows the limiting axial profiles. In general, the end effect is negative at a low burnup and increases as the burnup increases. At a low burnup, the neutron importance of the fuel mid-region, where reactivity is over-predicted, is greater since the flux shape is close to a cosine. This results in a negative end effect. At a high burnup, however, the flux shape significantly deviates

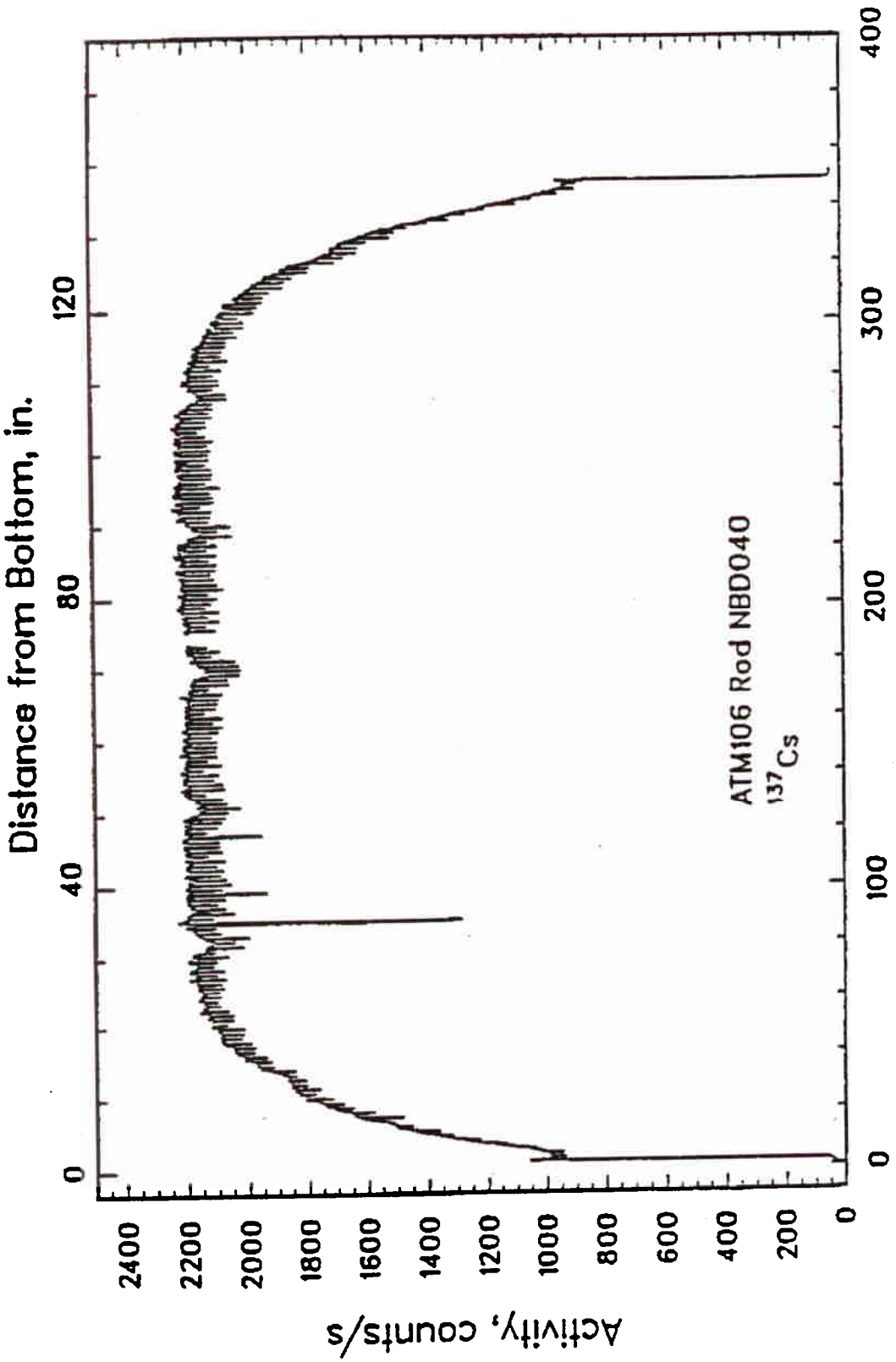
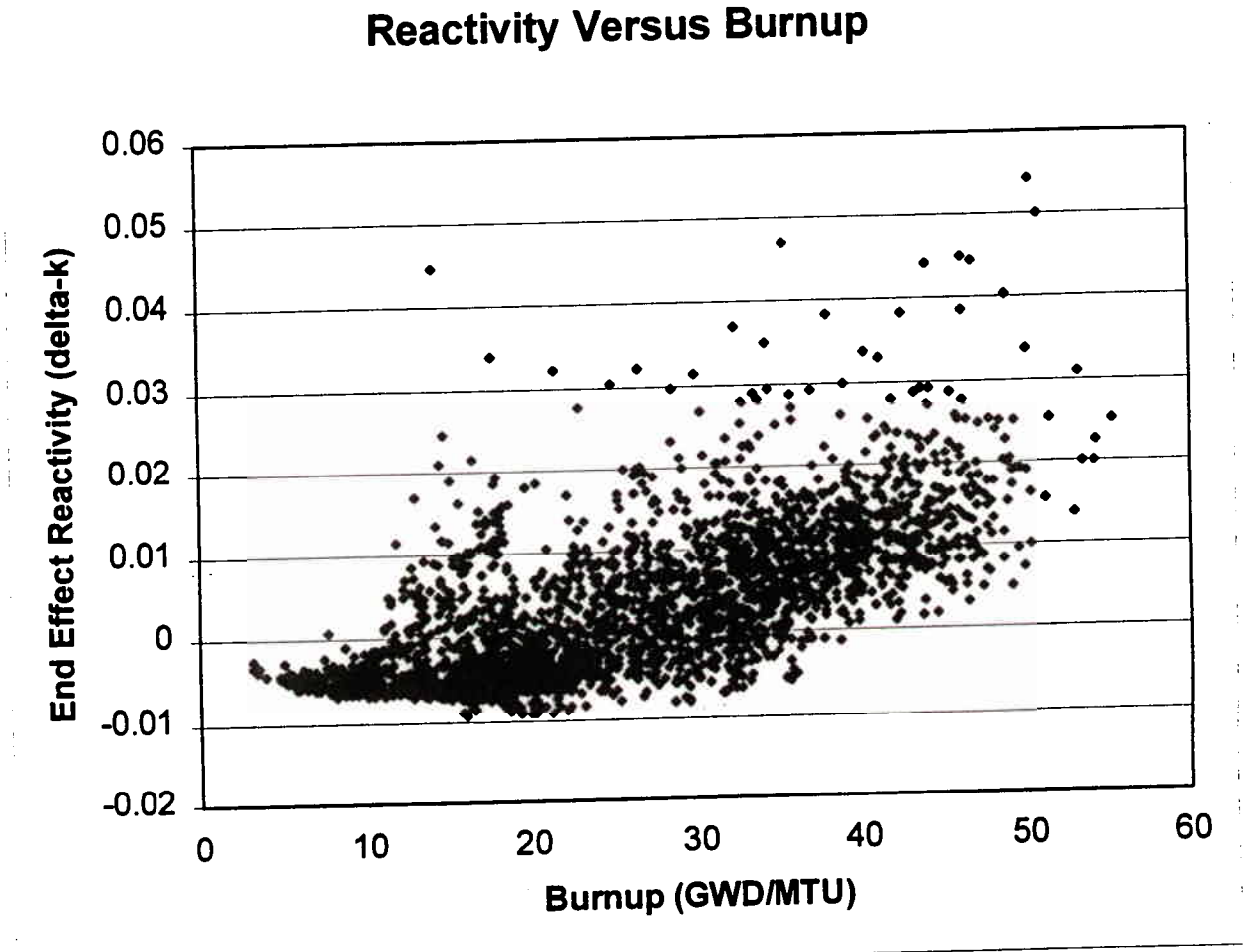


Figure 4-5. Burnup Profile Measurement by Gamma Scan

from the cosine shape and become more pronounced in the fuel end region. Thus, the fuel end region, where reactivity is under-predicted, becomes more important and the end effect becomes positive.

Figure 4-6. End Effect Reactivity versus Burnup



4.2.3.2 Axial Modeling Requirements

Any cask design using actinide-only burnup credit shall model the axial burnup with the appropriate 18 normalized, equal-size burnup, profiles presented in Table 4-3. Different profiles are to be applied depending on the assembly average burnup value. For example, an assembly with an average burnup of 25 GWd/MTU is to be analyzed with profile 2. Every analysis is to be performed based on the actual cask configuration with a chosen assembly type and cooling time. However, for the method described in this report, if a cask has an axially-varying poison plate design, the minimum poison concentration is to be assumed for the whole length. Also, if fuel assemblies can be shifted axially within a basket so that the entire active fuel length is not covered by poison plates, no credit is to be taken for their neutron-absorbing capability. Further,

if a fuel assembly employs multiple axial enrichment design, the maximum enrichment is to be assumed for the entire length. The same depletion code and the cross section library used for isotopic validation (Chapter 2) must be used for the calculation of actinide concentrations. The isotopic correction factors determined, consistent with the methodology presented in Chapter 2, must also be applied. The same criticality code and the cross section library used for criticality validation (Chapter 3) must be used with both the uniform and 18-node analysis.

Table 4-3. Limiting Axial Burnup Profiles

Axial Position (% of Core Height)	Normalized Burnup (Fraction of Assembly Average)		
	Profile 1 BU < 18 (GWd/MTU)	Profile 2 18 ≤BU < 30 (GWd/MTU)	Profile 3 30 ≤ BU (GWd/MTU)
2.78	0.649	0.668	0.652
8.33	1.044	1.034	0.967
13.89	1.208	1.150	1.074
19.44	1.215	1.094	1.103
25.00	1.214	1.053	1.108
30.56	1.208	1.048	1.106
36.11	1.197	1.064	1.102
41.67	1.189	1.095	1.097
47.22	1.188	1.121	1.094
52.78	1.192	1.135	1.094
58.33	1.195	1.140	1.095
63.89	1.190	1.138	1.096
69.44	1.156	1.130	1.095
75.00	1.022	1.106	1.086
80.56	0.756	1.049	1.059
86.11	0.614	0.933	0.971
91.67	0.481	0.669	0.738
97.22	0.284	0.373	0.462

4.2.3.3 Fission Source Convergence

The requirement of axially burnup-dependent analyses gives rise to a concern regarding fission source convergence in each burnup zone. Fission source convergence problems may result in non-conservative multiplication factors, and consequently incorrect loading curves. Thus, all 18 burnup zones must be adequately sampled. Due to the burnup profiles presented in Table 4-3 combined with the requirement of axially uniform poison, the axial fission source distribution should be confirmed as peaked in the top half of the cask.

An upper bound on the fission source convergence effects on multiplication factors can be inferred from OECD/NEA Burnup Credit Criticality Benchmark Phase IIA and IIB [4-7, 4-8]. A criticality benchmark problem of a given axial burnup profile has been analyzed by many different participating organizations using various code systems and neutron libraries. Although the isotopic compositions were provided, no restriction was placed on either eigenvalue or fission source convergence. Therefore, any variation in the reported multiplication factors is due to differences in code systems, neutron libraries, and convergence. The maximum standard deviation around mean multiplication values is 0.875% in k_{eff}. Even if the variation is assumed entirely due to problems in convergence, reserving 1% in k_{eff} from the fission product margins should be a conservative estimate. Although realistic effects should be considerably less, 1% in k_{eff} is reserved for fission source convergence problems. This value is tallied up against available fission product margins in Chapter 7. Using the fission product margin in this way removes any requirement for code-to-code comparisons. Future efforts in code-to-code comparisons could reduce the reliance on fission product margin.

4.2.3.4 Summary of Axial Burnup Profile

Axially burnup-dependent analyses are applicable only to those casks with an axially uniform poison concentration. If a cask employs axially varying poison plate design, the minimum poison concentration is to be assumed for the whole length. In addition, if fuel assemblies can be shifted axially within a basket so that the entire active fuel length is not covered by poison plates, no credit is to be taken for their neutron-absorbing capability. Further, if a fuel assembly employs multiple axial enrichment design, the maximum enrichment is to be assumed for the entire assembly. 1% in k_{eff} is reserved from the fission product margins for fission source convergence. Assemblies with part-length burnable absorbers are included from the viewpoint of the end effect because they are inserted to flatten the flux distribution. The flattened flux distribution eventually results in flattened burnup distribution and ultimately reduces the end effect. Part-length control rods are designed to perform a similar function and included from the end effect viewpoint. The limiting axial profiles shown in Table 4-3 are determined from a database which includes a number of assemblies irradiated with axial power shaping rods. Thus, the database and the burnup profile analysis properly reflect the effect of axial power shaping rods.

4.2.4 Horizontal Burnup Profile

Since the flux decreases near the core baffle, significant horizontal variation in burnup can exist in individual PWR assemblies, particularly if they are discharged following a single irradiation

cycle. Limiting arrangement of two or more assemblies with low burnup zones placed inward and adjacent to one another could potentially result in a reactivity increase in an SNF cask. This consideration is of special concern for small SNF cask designs where radial neutron leakage is significant, and thus, the orientation of fuel assemblies could make a significant change in the multiplication factor.

Figure 4-7 shows the maximum assembly quadrant deviation from the assembly average burnup with respect to the assembly averaged burnup determined from a compiled horizontal burnup database.[4-6] The horizontal burnup gradient is inversely proportional to the assembly averaged burnup, reflecting typical fuel management practices of moving assemblies from cycle to cycle to minimize the local power peaking and maximize the fuel economy. It is clear from Figure 4-7 that the values given in Table 4-4 conservatively estimate the horizontal burnup gradient expected in PWR assemblies. Any cask design utilizing actinide-only burnup credit shall use the values listed in Table 4-4. These values represent horizontal burnup gradient within a single fuel assembly. For example, an assembly with an average assembly burnup of 15 GWd/MTU is to be analyzed with 10 GWd/MTU (33% lower) on one half and 20 GWd/MTU (33% higher) on the other half, representing 33% deviation on each half. The most reactive loading configuration of multiple fuel assemblies must be identified by cask designers for their particular casks.

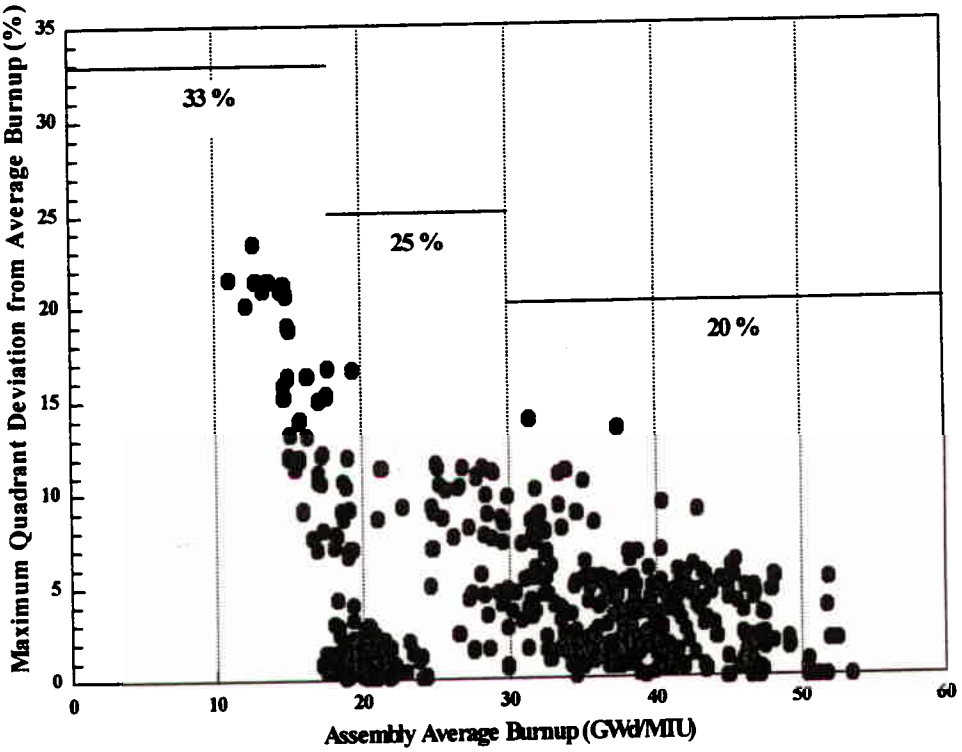


Figure 4-7. Maximum Quadrant Deviation versus Assembly Average Burnup[4-6]

Table 4-4. Conservative Horizontal Burnup Gradients in PWR Assemblies

Assembly Average Burnup (GWd/MTU)	Horizontal Gradient (%)
< 18	33
18≤ and < 30	25
30 ≤	20

4.3 CONCLUSIONS

This chapter defined the limiting parameters for the isotopic depletion analyses and the limiting SNF package analyses that must be performed for criticality calculations. This topical report seeks NRC acceptance of the limiting values for these parameters as presented in Table 4-5 below.

Table 4-5. Limiting Values of Modeling Parameters

Parameter	Analysis Affected	Value/Assumption
Cycle History	Depletion	One Irradiation Cycle (No Downtime)
Specific Power	Depletion	60 MW/MTU
Moderator Density (In Reactor)	Depletion	Maximum Average Core Outlet Temperature
Dissolved Boron	Depletion	Maximum Cycle Average ppmb
Fuel Pellet Temperature (In Reactor)	Depletion	Maximum Average Pellet Temperature
Moderator Density (In SNF Package)	Criticality	Search for Maximum Reactivity
Fuel Pellet Temperature (In SNF Package)	Criticality	Ambient Temperature 20 °C (293 K)
Axial Burnup Profile	Criticality	Axially Burnup-Dependent, 18-Node Analysis with Profiles in Table 4-3
Horizontal Burnup Gradient	Criticality	Horizontal Burnup Gradients in Table 4-4

5. LOADING CRITERIA

Burnup credit loading curves are the criteria used to determine whether it is permissible to load an assembly in an SNF package using burnup credit. This chapter describes the steps required to develop burnup credit loading curves. These curves identify the lowest acceptable burnup as a function of the initial enrichment. To generate a loading curve, the maximum fresh fuel enrichment meeting the upper safety limit on k_{eff} is determined. Subsequently, a curve of required minimum burnup versus initial enrichment is developed by applying the burnup credit methodology at various initial enrichments. Loading curves may be developed for each assembly type which will be put in the SNF package. Since additional cooling time makes the loading curves less restrictive, the loading curves can also be generated as a function of cooling time. In general, there will be a single loading curve applicable to each specific combination of cask design, assembly type, and assembly minimum cooling time.

5.1 FRESH FUEL CALCULATIONS

The maximum fresh fuel U-235 enrichment that may be used in a given SNF package is determined first. The k_{eff} is calculated with a validated code system (Chapter 3) for a range of initial enrichments to determine the enrichment that produces a k_{eff} (or $k + 1.645\sigma$ for Monte Carlo results) equal to the upper safety limit. This is the maximum fresh fuel enrichment point and is labeled as (E_4, 0) on the loading curve (Figure 5-3). The loading curve consists of an abscissa that represents initial (fresh) fuel enrichment and an ordinate that represents the required minimum burnup for a given initial enrichment. Next, a vertical line is drawn at the maximum fresh fuel enrichment limit. All assemblies that have initial U-235 enrichments less than or equal to the maximum fresh fuel enrichment limit, E_4, may be stored or transported regardless of burnup.

5.2 GENERATION OF THE BURNUP CREDIT LOADING CURVE

5.2.1 Find the Limiting Burnup for Each Initial Enrichment

The required minimum burnup for a specific initial enrichment value is the burnup at which the calculated k_{eff} (or $k + 1.645\sigma$), using the burnup credit methodology, is just equal to the upper safety limit. The process for determining a required minimum burnup for a given initial enrichment is illustrated in Figure 5-1. A series of runs of validated computer codes (i.e., SAS2H and CSAS25) is performed within each axial burnup profile range to calculate k_{eff} values for a range of burnups to search for the burnup value that produces the reactivity limit. The reactivity limit is the upper safety limit as determined in Chapter 3. As indicated in Figure 5-1, the calculated k_{eff} is plotted against the burnup that produced that value of k_{eff}. The curve is then fit to estimate the burnup that crosses the upper safety limit. The process is repeated for various initial enrichments as illustrated in Figure 5-1. A calculation is performed near that burnup (for each initial enrichment value) which will be less than or equal to the upper safety limit. This limiting burnup will be used with the corresponding initial enrichment to establish a point on the burnup credit loading curve.

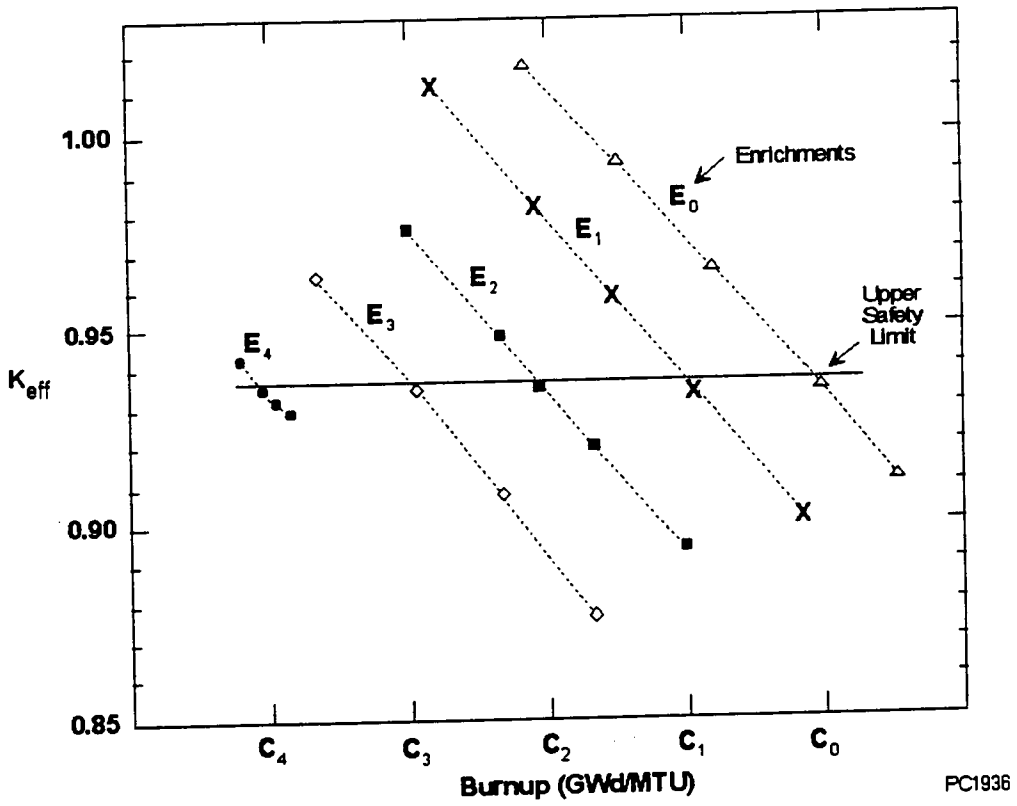


Figure 5-1. Determination of Required Minimum Burnups for a Specific SNF Package

5.2.2 Find the Limiting Initial Enrichment at Burnup Discontinuities

The loading curve will contain discontinuities at 18 and 30 GWd/MTU. These are due to changing the axial and horizontal burnup models at these values. The process of determining required minimum initial enrichments at the burnup discontinuities is shown in Figure 5-2. The process is similar to the description in Section 5.2.1. However, the initial enrichment is varied this time while the burnup is fixed at 18 and 30. Two different minimum initial enrichments result, depending upon whether the lower-burnup or upper-burnup axial burnup profile and horizontal burnup gradient is selected at the 18 and 30 GWd/MTU interface points. It is required that the lower of the two minimum initial enrichments be determined. It has been shown (Table 5-6 of Reference 4-2, with profiles 1, 2, and 3 in this topical report corresponding to profiles 9, 8, and 5, respectively, in the fore mentioned reference) that the use of the lower-burnup axial profiles 1 and 2 at the 18 and 30 GWd/MTU interfaces, respectively, are more reactive. Similarly, the lower-burnup horizontal burnup gradients at the interfaces are more reactive because the effect is proportional to the burnup gradient, which is larger for the lower-burnup side of the interface. Thus, the lower of the two minimum enrichments at each interface are determined by using the axial burnup profile 1 in Table 4-3 and the horizontal burnup gradient of 33% to determine the minimum initial enrichment for the 18 GWd/MTU interface point, and axial burnup Profile 2 in Table 4-3 and the horizontal burnup gradient of 25% to determine the minimum initial enrichment for the 30 GWd/MTU interface point. Determining the other initial enrichment is optional, as discussed in the following paragraph. This can be accomplished by using axial burnup profiles 2 and 3 in Table 4-3, and the horizontal burnup gradients of 25% and 20% for 18 and 30 GWd/MTU, respectively. The distance between E_5 and E_6 or E_7 and E_8 is expected to be on the order of 0.2 or 0.1 wt. % U-235, respectively.

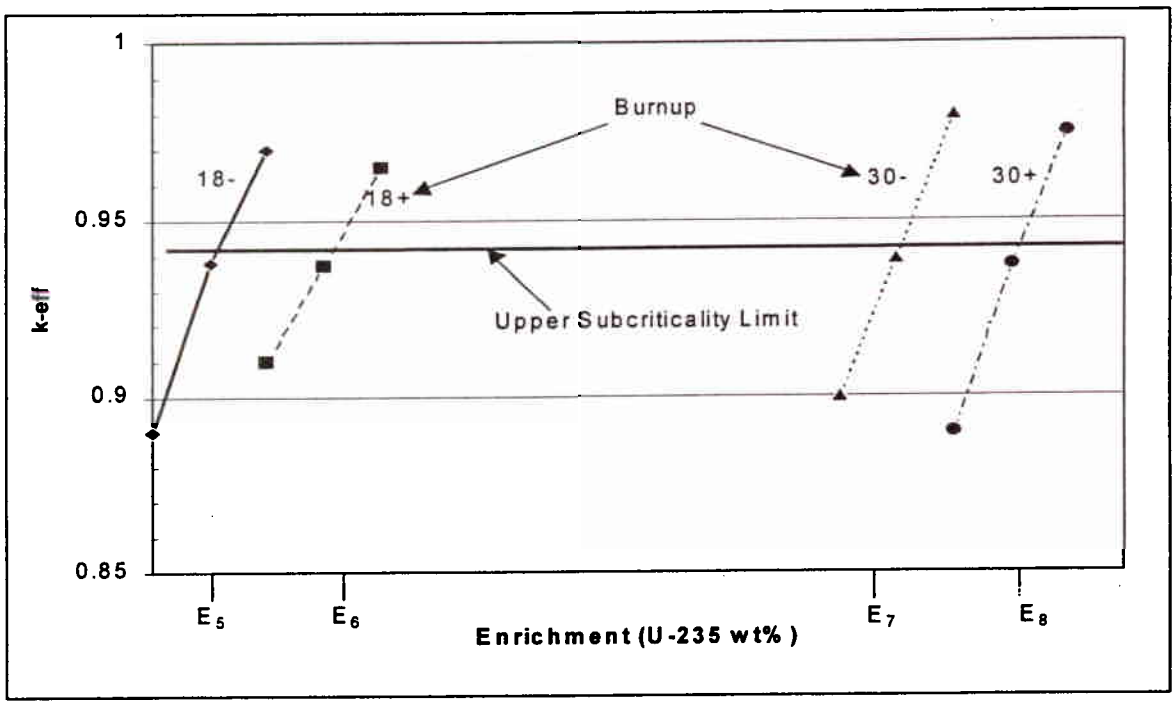


Figure 5-2. Determination of Limiting Initial Enrichment at Burnup Discontinuities

5.2.3 Plot the Burnup Credit Loading Curve

After the calculations of Sections 5.2.1 and 5.2.2 are performed, a curve of minimum burnup as a function of the initial enrichment is generated (see Figure 5-3). Calculations of the required minimum burnup must be performed at the maximum enrichment for the SNF package (E_0). (This limit is often not set by burnup credit concerns. The limiting enrichment for this burnup credit methodology is 5 weight percent U-235.) Calculations of the required minimum burnup must also be performed at the maximum fresh fuel enrichment for the package (E_4). Burnup credit calculations will not show a zero minimum burnup for the maximum fresh fuel limit demonstrated using fresh fuel assumptions. This is because in performing the calculations, the isotopic correction factors on U-238 and U-235 are used that only need to be applied for irradiated fuel. The required minimum burnup for the highest enrichment is indicated as point C_0 on Figure 5-3. Subsequent values C_1 through C_n are obtained by decreasing the initial enrichment parameter by a value not to exceed 0.5 weight percent U-235 until an initial enrichment equal to the maximum fresh fuel enrichment limit is reached. The optimum moderation must be checked at point (E_4, 0) and the point (E_0, C_0). If the optimum moderator density is not 1.0 at both points, then optimum moderation must be found for each point. The required minimum initial enrichments, E_5 and E_7, must be found at 18 and 30 GWd/MTU. The loading curve is created by a segmented straight line through the data points. Points (E_6, 18) and (E_8, 30) may be determined and incorporated into the loading curve, but this is optional. If there is significant curvature in the loading curve at burnups other than 18 and 30 GWd/MTU, the enrichment points should be spaced so that the loading curve is smooth, with no abrupt direction changes.

The loading curve is based on the assured minimum assembly-average burnup. Because utilities are responsible for cask loading and also for assembly burnup data, the procedures for providing assured assembly burnups are also the responsibility of the utilities, and are not the subject of this topical report. However, those procedures, which are briefly outlined in Section 6, are expected to consist of using reactor burnup records, with independent burnup confirmation, and subtracting the amount of the burnup uncertainty in the assembly's burnup. A spent fuel assembly that has an assured average burnup greater than the required minimum burnup on the loading curve, at the assembly's initial enrichment, may be loaded into the SNF package. Note that an assembly that has an initial enrichment less than the maximum fresh fuel enrichment limit does not require any burnup. Conversely, an assembly that has an initial enrichment that exceeds the highest enrichment on the loading curve may not be loaded into the package regardless of its burnup. If an assembly is initially loaded with fuel of different enrichments, the maximum enrichment value at any point in the assembly is used for the assembly in comparing the assembly to the loading curve. This conservatively bounds the reactivity of such an assembly.

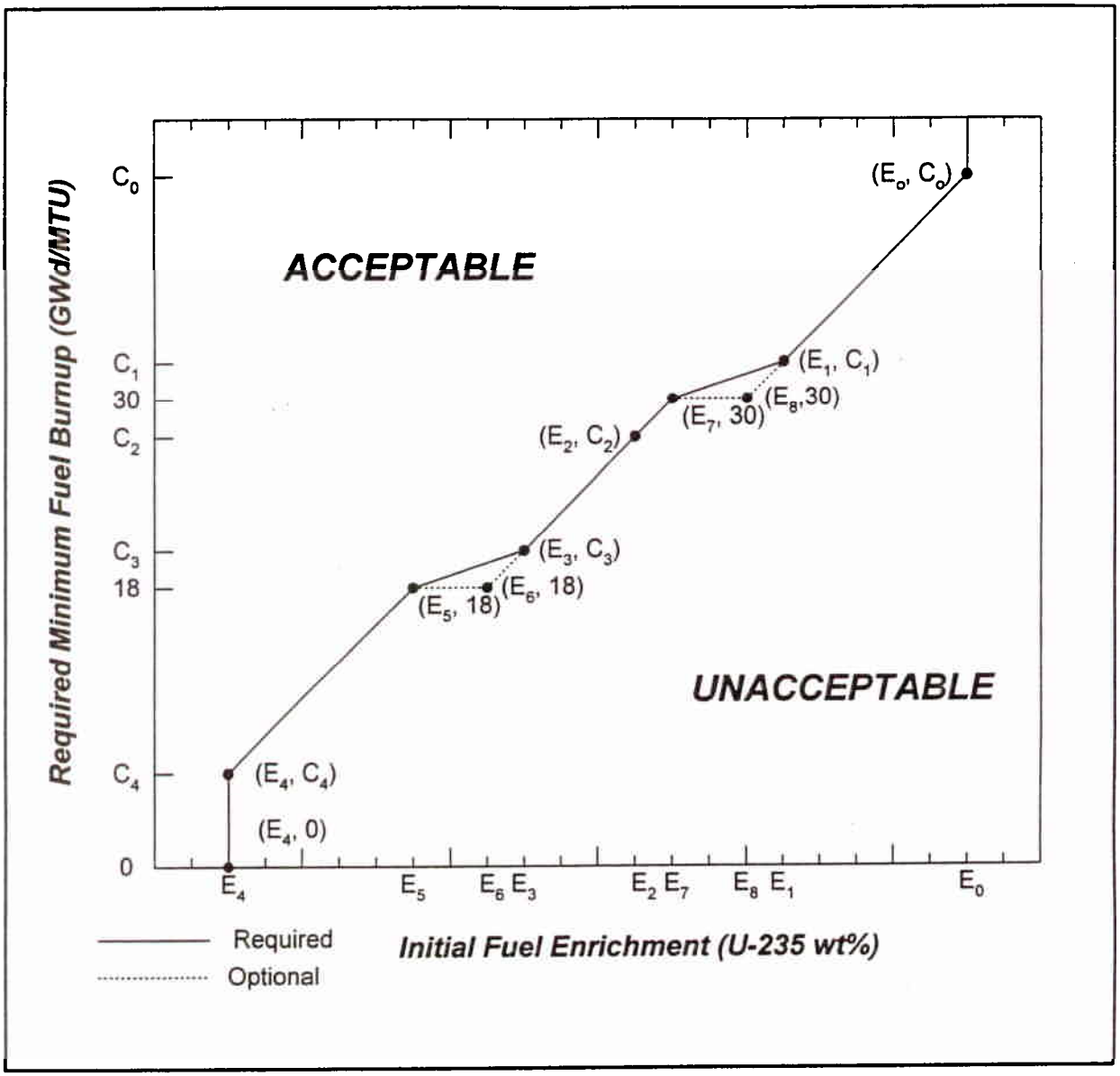


Figure 5-3. Development of the Burnup Credit Loading Curve for a Specific SNF Package

5.3 LIMITATIONS ON THE BURNUP CREDIT LOADING CURVES

A burnup credit loading curve will be valid for a class of assemblies. The class is characterized by the assembly design type, the number of removable burnable absorbers (if any) used in the assembly, and cooling time. The limitations on the acceptable parameters for a class of assemblies for each loading curve shall be notated on the curve as illustrated in Figure 5-4. There are also parameters that are not intended to identify a class, but to be generically acceptable (maximum cycle average ppm boron, maximum core outlet temperature, and maximum pellet average temperature), which are notated on the loading curve in case unanticipated design changes invalidate the generic assumptions. The cask vendor will decide on which loading curves will be included in his license application. He would normally include at least one loading curve for the fuel types of each of his current utility customers. He would choose a cooling time, the appropriate number of removable burnable poison absorbers, and other parameters, based on his customers' fuel characteristics. Additional load curves could be included where these would increase the fraction of a customer's fuel that could be accommodated by his cask.

The following subsections describe the parameters that can be varied for which a separate loading curve is able to be generated in lieu of establishing a bounding curve.

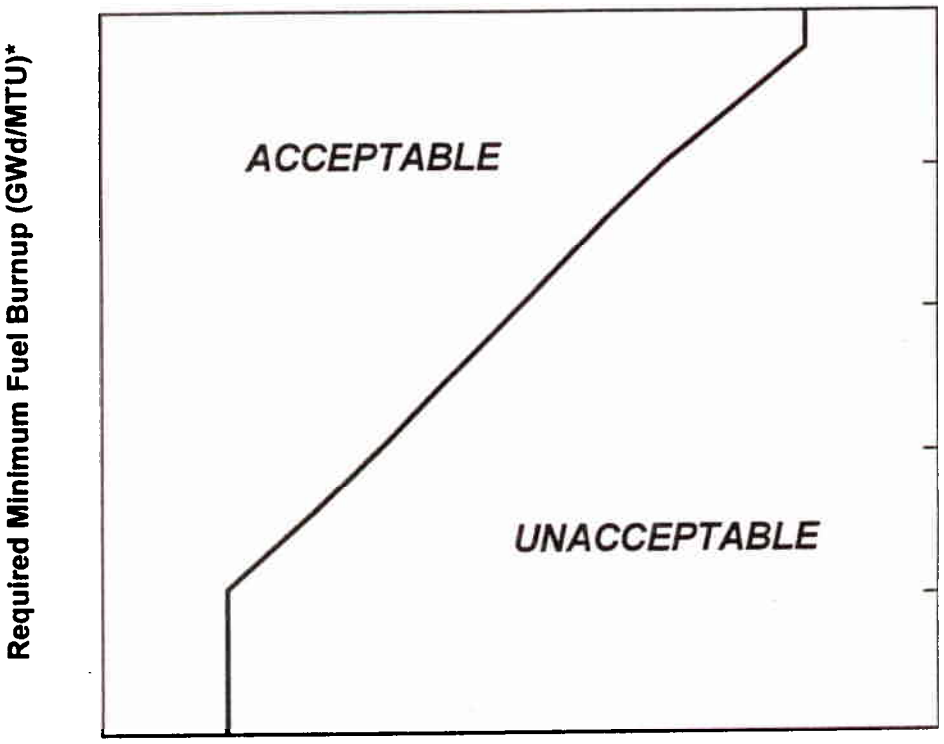
5.3.1 Assembly Design Type

PWR fuel assembly designs vary in their rate of change in reactivity with burnup. Typically, a design that has a higher hydrogen to uranium ratio (H/U ratio) will initially have a higher reactivity for a given enrichment. This high H/U design, however, will typically lose more reactivity for a given burnup than a low H/U design. With this observation, it is clear that there is no one assembly design that would be the most limiting at all burnups on a burnup loading curve. In addition, the assembly type that has the highest reactivity in the cask may be dependent upon the specific design of the spent fuel basket.

The consequence of this variability in assembly design is that separate burnup credit loading curves will be generated for each fuel assembly design type. Assemblies with fixed burnable absorbers represent an assembly design type. Assemblies with more than one fuel enrichment must be analyzed as though they have a uniform enrichment, with that enrichment being the highest in the assembly. Multiple enrichments cannot be represented as a separate assembly design type.

5.3.2 Assemblies Loaded With Removable Burnable Absorber Rods

The insertion of burnable absorber rods into a fuel assembly for a cycle affects the irradiated fuel isotopic composition by hardening the neutron spectrum. This hardened spectrum results in more U-238 fast fission and a higher conversion ratio. The net effect is that the fuel assembly isotopic composition and reactivity characteristics as functions of burnup deviate from those for assemblies without burnable absorbers. The assemblies that contained burnable absorbers will have a higher reactivity for a given burnup and enrichment than those that did not. The effect increases with larger



Assembly Design: _____
Minimum Cooling Time: _____
Maximum Number of Removable Burnable Absorber Rods _____

Note: This loading curve was generated with the following generic assumptions:
Maximum Cycle Average ppm Boron of_____. Maximum Core Outlet Temperature of _____,
and the Maximum Pellet Average Temperature_____

*The nominal burnup must be reduced by the utility so there is a 95% confidence level of meeting the Required Minimum Fuel Burnup.
**If the assembly has more than one enrichment, the highest enrichment must be used.

Figure 5-4. Burnup Credit Loading Curve

amounts of burnable absorbers in the assembly. This effect is generally small but may be as large as a few percent in reactivity.[5-1]

Separate burnup credit loading curves or a bounding treatment of burnable absorber rod effects must be included for each reactor fuel design covered by an SNF package design Safety Analysis Report. A burnup credit loading curve will state whether it applies to fuel with burnable absorbers. Typically, burnable absorber assemblies are removed after one cycle. However, the SNF depletion analysis will be performed with the burnable absorbers in the assembly throughout the life of the assembly to bound the possible time actually in the assembly. In the criticality analysis for the package, the depleted burnable absorbers will not be modeled. This is a conservative assumption for all fuel designs. The more burnable absorber rods assumed in the isotope depletion/generation calculations, the larger the positive reactivity effect. Due to this, it is conservative to perform the analysis with the maximum burnable absorber loading during operation in the reactor. Loading curves developed with burnable absorbers could be conservatively applied for fuel without burnable absorbers. Reactor records provide the necessary documentation to determine whether an assembly had a burnable absorber loaded any time during exposure in the core. Verification of assembly records is addressed in the next chapter.

5.3.3 Cooling Time

The cooling time after discharge of an SNF assembly from the reactor affects the isotopic inventory within the fuel material since many isotopes are unstable and decay with time. A study of the k_∞ of spent fuel versus cooling time with several operating history options was performed.[5-2] Figure 5-5 shows that shortly after discharge from the reactor, the reactivity decreases monotonically for the first 100 years. The decrease in k_∞ for the actinide case is mainly due to decay of fissile Pu-241, which has a half life of 14.4 years. It is also noted that the increase in reactivity after 100 years is very gradual, and that the reactivity at 200 years remains well below the reactivity at 25 years of cooling. This gradual reactivity increase is due to the decay of Pu-240 and Am-241. The negative reactivity worth of fission products increases with cooling time; therefore, neglecting fission products adds more conservatism with cooling time.

Since reactivity decreases with cooling time during the first 25 years, and remains below the 25-year level for at least 200 years, a cask designed for 25 years or less cooling time would be capable of transporting fuel with cooling times up to at least 200 years. Thus, a loading curve based on 25 years or less cooling time would be valid for any cooling time up to 200 years. Therefore, the scope of this topical report is limited to 200 years of cooling time, with the additional requirement that the maximum cooling time that can be used to develop loading curves is 25 years. The specific cooling time used to develop a loading curve must be stated on the loading curve.

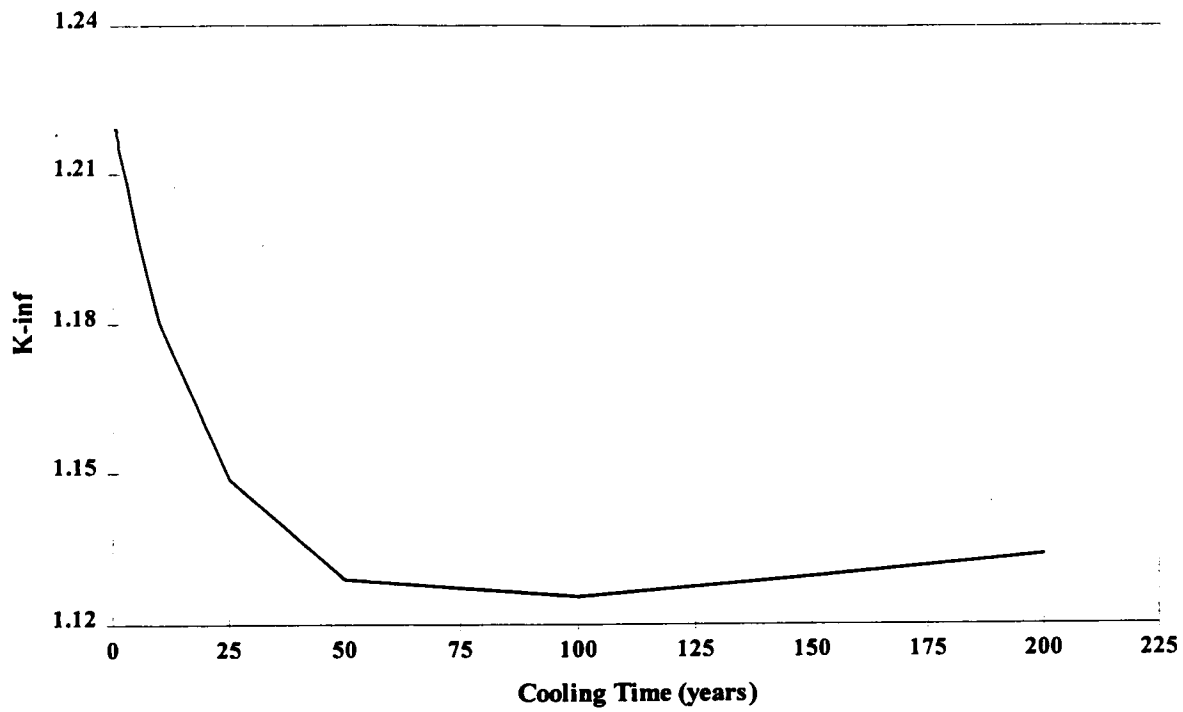


Figure 5-5. k_{inf} versus Cooling Time (Actinides only, 3.0 wt.% U-235, 30 GWd/MTU)[5-2]

5.4 SUMMARY

Burnup credit loading curves are generated that establish the minimum burnup that can be loaded into an SNF package as a function of initial enrichment. These curves are generated using the conservative isotopic correction factors presented in Chapter 2, the upper safety limit developed in Chapter 3, and the conservative burnup analysis presented in Chapter 4. The package criticality analysis is described in Chapter 4. The loading curve thus includes a methodology-specified allowance for uncertainties in the designer's criticality design process. The reactor record burnup uncertainty is accounted for by utilities following guidelines that are being developed, guidelines that will be the subject of separate, future NRC review.

Burnup credit loading curves are generated for each assembly and SNF package design. The use of burnable absorbers may be considered as a separate design. The more burnable absorbers, the more reactive the assemblies. Therefore, a loading curve can be valid for any assembly with fewer burnable absorbers installed during plant operation than that used for the analysis. The burnup credit loading curve will be calculated with an assumed minimum cooling time. An assembly with a cooling time greater than the burnup credit loading curve design basis cooling time, but less than 200 years, may be loaded. The limits of the burnup loading curve will be indicated on the loading curve, and the assemblies to be loaded will be verified to meet these limits.

6. PHYSICAL IMPLEMENTATION AND CONTROLS

NRC approval of the burnup credit methodology described in this topical report is expected to result in the design, licensing, certification, and fabrication of burnup credit transport casks and transportable storage canisters with transport overpacks. When these burnup credit packages are actually loaded, the utility user will both perform the physical loading process and provide the reactor and fuel data and the evaluations which support the loading, including the assured compliance with the package's loading curves. Because these important implementation functions are performed by the individual utility user, and are still under development, the physical implementation and control processes for the loading of burnup credit transport packages are not included in the scope of this topical report.

The purpose of including this section and its discussion is to demonstrate that there are several reasonable and appropriate implementation procedures available. These are expected to lead to successful utility utilization of the cask loading curves that are the result of the approved topical methodology and its application in the design and certification of burnup credit transport packages. The principal utility implementation functions described below are:

- Verification that each assembly is appropriate to the loading curve being used
- Development of the minimum assured assembly-average burnup that will be used with the loading curve
- Assurance against fuel assembly misload.

6.1 VERIFICATION OF ASSEMBLY SUITABILITY

Specific burnup credit package loading limits are established in the Certificate of Compliance or Safety Evaluation Report and are discussed in the Safety Analysis Report for the specific package design. As addressed in previous chapters, the parameters to be used in establishing the loading limits for a burnup credit package include the fuel assembly type, initial enrichment, assembly average burnup, burnable absorber irradiation history, and cooling time.

Burnup credit loading curves and their related limits provide the operational limits for selecting fuel assemblies for loading into a burnup credit package. Physical implementation of burnup credit involves utility operational preparations, including development and implementation of fuel classification procedures for actual fuel loading operations. Fuel assembly classification procedures will be prepared before loading operations and will be unique to specific package designs and to individual utility circumstances. Reactor records and the burnup credit package loading criteria are used to segregate spent fuel assemblies as acceptable or not acceptable for loading into a burnup credit package. Prior to commencement of burnup credit package loading operations, these procedures are implemented, using reactor records, to identify spent fuel assemblies that do not meet the applicable loading criteria specified for the burnup credit package. These assemblies are excluded from further consideration for loading into the package. The

remaining assemblies that do meet the criteria, and whose average burnups exceed the load curve minimum burnup appropriate for their initial enrichment, are then evaluated with respect to the remaining package loading criteria unrelated to criticality safety. All fuel assemblies that meet all applicable loading criteria are considered to be "candidate" fuel assemblies for loading into the specified package design.

6.2 DEVELOPMENT OF MINIMUM ASSURED ASSEMBLY BURNUPS

The loading of a qualified assembly into a certified transport package requires that the assured assembly average burnup be no less than the load curve burnup for that assembly's enrichment. The assurance of the assembly burnup has two basic requirements: the requirement for an independent direct measurement of the assembly's burnup; and an allowance, in the form of a burnup reduction, for the uncertainty in the assembly burnup. The instrumentation and methodology of direct burnup measurement involve site-specific circumstances, which need to be addressed by the individual users. The nature of assembly burnup uncertainty in reactor records is being investigated by the utilities, and the uncertainty in burnup measurements will be a characteristic of the instrumentation and methodology being used.

There appear to be at least two alternative ways of using reactor records of assembly burnup, direct assembly burnup measurements, and assembly burnup uncertainty data for quantifying the assured assembly-average burnup to be used with the load curve. Each of the two methods discussed below relies on a calibration of the measurement system by measuring a large number of assemblies, and using reactor record burnup data for these same assemblies to establish the calibration. The two alternative methods are:

- Use the reactor record assembly burnup, when confirmed to be within a predefined range by a direct assembly burnup measurement, reduced by the uncertainty in the reactor burnup record. Methods of determining the uncertainty in the reactor records are being developed.

- Use the measured assembly burnup, as calibrated by the reactor record burnup data, reduced by the uncertainty in the measured burnup, as determined statistically from all of the measurements in the calibration.

The use of the first method, above, has the advantage of operational simplicity, in which the assured assembly burnup can be determined prior to measurement, from reactor data, and be used subject only to the measurement-based confirmation of the reactor record assembly burnup. The second method has the advantage that both the burnup and its uncertainty are measurement-based. The existence of at least two potential methods of establishing assured assembly burnups for use with loading curves gives reasonable assurance that one or more such methods can become a part of acceptable regulatory practice for use during the loading of certified transport packages.

6.3 ASSURANCE AGAINST FUEL ASSEMBLY MISLOAD

Operational fuel handling and control procedures and practices that provide assurance against fuel assembly misload have been approved and successfully used to verify the correct fuel loadings in reactor cores, in reactor spent fuel pool storage racks, and in dry storage casks. These same procedures, when adapted to the loading of burnup credit transport packages, are anticipated to provide controls over nuclear criticality safety practices that satisfy the Double Contingency Principle of ANSI/ANS-8.1.[6-1] Such an adaptation could be as follows: Assembly identifiers are independently verified by two fuel handling operators at each stage of the burnup verification and package loading procedures, and the reactor record burnup levels are independently verified using a measurement system prior to spent fuel package loading. These procedural measures ensure proper assembly selection and records assignment. Therefore, the loading procedures incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent errors in the loading process to occur before a criticality accident is possible. This conclusion does not rely on PWR storage pool soluble boron credit and is valid assuming pure water moderation as a normal preexisting condition.

6.4 SUMMARY

This section has discussed the physical implementation and control processes for the loading of certified burnup credit transport packages. These processes will be the conducted by the utility that loads such packages, and are therefore not a part of the scope of this topical report. Alternatively, the discussion in this section is intended to demonstrate that there are reasonable and, in many cases, established implementation procedures and alternatives available. It is therefore expected that acceptable processes and procedures will be available as the necessary implementation complement to an approved burnup credit methodology and the design, certification and availability of burnup credit transport packages.

7. SUMMARY OF THE BURNUP CREDIT PROCESS

This chapter summarizes the burnup credit methodology presented in the previous chapters. It provides a review of the burnup credit process, a discussion of the range of applicability, conservatism in the methodology, and a summary of the NRC approvals sought.

7.1 OVERVIEW

The burnup credit process was introduced in Chapter 1, Figure 1-2. The process builds upon those currently used for cask analysis and operations (i.e., the fresh fuel assumption and reliance on utility records for assembly initial enrichment). These steps are supplemented by analytical steps and operational procedures that are unique to burnup credit.

The fresh fuel assumption considers a cask loaded only with unirradiated (fresh) fuel assemblies. With that assumption, the initial enrichment of the fuel assembly is the single parameter upon which cask criticality safety is based. The cask design is analytically shown to satisfy the criticality safety criterion, i.e., $k_{eff} \leq 0.95$ including all bias and uncertainties, for specified fuel designs with initial enrichments less than the design basis enrichment. Operationally, reactor records for assembly initial enrichment are used to qualify assemblies to be loaded into the cask.

Burnup credit adds fuel assembly burnup as a second key qualification parameter for criticality safety. This requires determining the reactivity relationship between the required fuel assembly burnup and initial enrichment over the range of allowable enrichments to establish loading criteria for the cask. The criteria are curves of burnup versus enrichment called the burnup credit loading curves. An example of a loading curve is shown in Figure 7-1. Over the range of enrichment, assemblies with burnups above and to the left of the curve (the acceptable region) may be loaded into the SNF package; those with burnups below and to the right of the curve (unacceptable region) may not. The utility then selects the assemblies for loading based on their enrichments and burnup, and verifies the proper assembly selection prior to declaring an assembly qualified for loading into a specific burnup credit package.

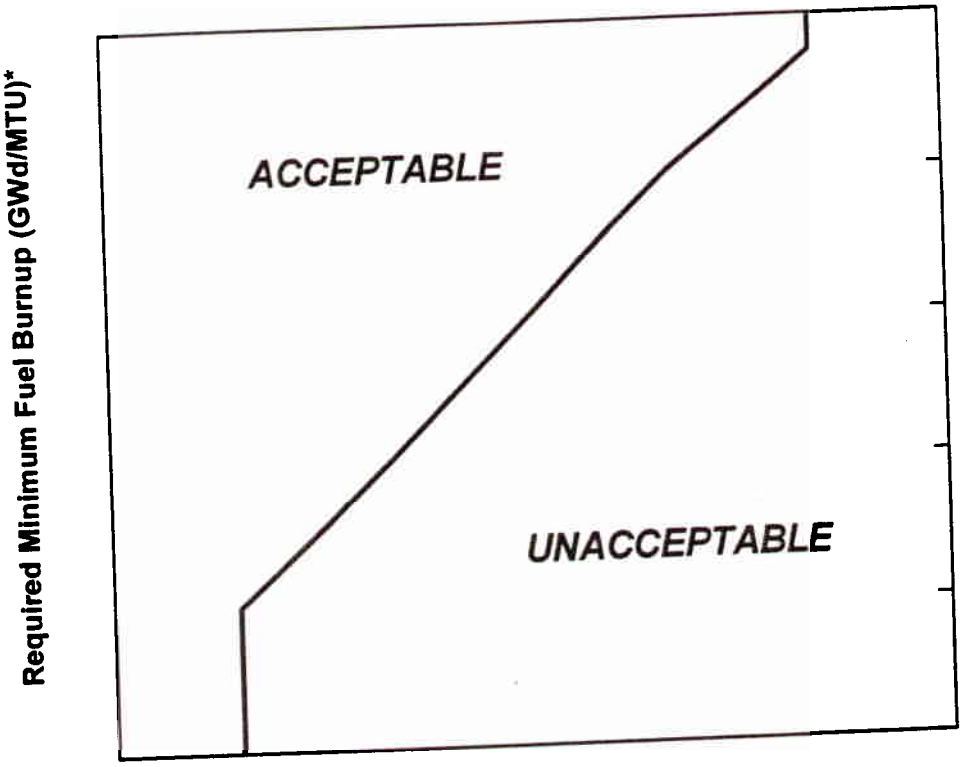
7.2 REVIEW OF THE BURNUP CREDIT STEPS

There are five major steps to implementing burnup credit:

1. Validate a computer code system to calculate isotopic concentrations in SNF created during burnup in the reactor core and subsequent decay.

2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff}, of a spent nuclear fuel package.

3. Establish bounding conditions for the isotopic concentration and criticality calculations.



Assembly Design: _____

Minimum Cooling Time: _____

Maximum Number of Removable Burnable Absorber Rods _____

Note: This loading curve was generated with the following generic assumptions: Maximum Cycle Average ppm Boron of_____. Maximum Core Outlet Temperature of _____, and the Maximum Pellet Average Temperature_____

*The nominal burnup must be reduced by the utility so there is a 95% confidence level of meeting the Required Minimum Fuel Burnup.

**If the assembly has more than one enrichment, the highest enrichment must be used.

Figure 7-1. Burnup Credit Loading Curve

4. Use the validated codes and bounding assumptions to generate package loading criteria (burnup credit loading curves).

5. Verify that SNF assemblies meet the package loading criteria and confirm proper assembly selection prior to loading.

It should be noted that steps one through four are to be performed by the package designer, while step five is the particular utility's responsibility. Table 7-1 summarizes the key steps in the burnup credit process. The following sections provide a brief description of each of these steps.

7.2.1 Validation of a Code System for Calculation of Isotopic Concentrations

This topical report addresses three separate steps in the validation of isotopic concentration predictions. First, a data set suitable for the validation is presented. Second, a method for the isotopic validation is developed, consisting of best estimate analyses of the data and then conservative biasing of the isotopic results. Finally, the data and method are demonstrated in validating a code system.

A set of chemical assays of spent nuclear fuel is presented in this topical report. The chemical assay data come from measurements of PWR fuel assemblies. The range of applicability for the measured data is discussed in Section 2.2.2. The set of experiments is sufficient for burnup credit analysis using actinides-only. This set of experiments could be used with any computer code system to validate burnup credit.

The method of analysis consists of conservatively selecting isotopes, followed by a method to determine biases, uncertainties, and then conservative correction factors. The isotopes selected are U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241. The biases are multiplicative and are the expected mean of the ratio of the experimental observation over the best estimate calculation of the isotopic concentration determined for each isotope. The uncertainty is determined by standard statistical procedures. The conservative correction factors are determined by combining the bias and uncertainties in a conservative direction for each isotope. (For example, if an isotope has a bias of 0.98 and an uncertainty of 5%, the isotopic concentration correction factor would be calculated as $0.98+0.05 = 1.03$ for a fissile material but $0.98-0.05 = 0.93$ for an absorber.) If an isotope shows a trend with burnup, burnup*spectrum, burnup*enrichment, or burnup*specific power, the mean bias and the uncertainty are treated as a function of that trending parameter.

Using the method and data presented in this topical report, the SAS2H module of SCALE 4.2 code system with the 27BURNUPLIB cross section set has been used in calculating the isotopic concentrations for the purpose of a demonstration. The biases, uncertainties, and correction factors used are presented in Chapter 2.

Table 7-1. Burnup Credit Analysis Process

Summary Section (Steps)	Step in Burnup Credit Process	Detailed Technical Basis Section
7.2.1	ISOTOPIC VALIDATION	
	Determine which isotopes to use in the analysis (currently restricted to selected actinides only).	2.1
	Perform best estimate analysis of isotopic concentrations of chemical assay measurements of spent nuclear fuel (validated for SCALE 4.2 in this report).	2.2
	Determine biases and uncertainties for each isotope and calculate conservative correction factors (validated for SCALE 4.2 in this report).	2.3
7.2.2	CRITICALITY VALIDATION	
	Perform best estimate analysis of the selected critical experiments.	3.1
	Establish bias and uncertainty.	3.2
	Establish the range of applicability.	3.3
	Establish an acceptance criteria.	3.4
7.2.3	LIMITING PARAMETERS	
	Determine highest moderator temperature and fuel temperature for all fuel assemblies to be put in the SNF package.	4.1
	Determine highest cycle average soluble boron concentration during burnup for any assembly to be put in the SNF package.	4.1
	Determine the moderator density that yields the highest k_{eff} for the SNF package criticality analysis.	4.2

Summary Section (Steps)	Step in Burnup Credit Process	Detailed Technical Basis Section
7.2.4	CONSTRUCTION OF LOADING CURVE	
	Use validated code to calculate the maximum fresh fuel enrichment that can be loaded in the SNF package.	5.1
	Use validated code and limiting values to compute spent fuel isotopic composition and apply the isotopic correction factors.	5.2
	Use validated code, limiting axial (or k_{eff} bias curves) and horizontal burnup profiles, and the limiting moderator density to compute k_{eff} for the package for an enrichment and burnup.	5.2
	Repeat the above two steps for a series of enrichments and burnups establishing limiting burnup for each enrichment where k_{eff} is less than or equal to the USL.	5.2
	Repeat this section's steps for each assembly design for a selected cooling time and burnable absorber loading.	5.3
7.2.5	LOADING VERIFICATION	
	Identify candidate assemblies satisfying the loading criteria and verify assembly IDs.	6.1
	Verify reactor records are consistent with the selected assembly characteristics for each assembly.	6.1

7.2.2 Validation of a Code System for Calculation of Criticality in an SNF Package

The "fresh fuel assumption" requires the cask vendors to determine an appropriate set of UO_2 critical experiments. This is still required for burnup credit. In conjunction, with this UO_2 critical experiment, 47 MOX critical experiments were selected to cover the plutonium isotopes and Am-241. The statistical treatment for determining biases and uncertainties resulting in an Upper Subcriticality Limit (USL) is performed in the same manner as for the fresh fuel assumption. The only additional requirement for actinide-only burnup credit is that USLs must be established from the UO_2 set selected by the vendor and the MOX set established in this report. The most limiting of the two USLs is taken as the limit. In the case where a trend against some parameter, such as a spectral index, is determined, it is possible that the limiting USL will come from a combination of the two USLs previously determined. Since the MOX set of experiments is limited, 2% in k_{eff} from the fission product margin has been reserved.

7.2.3 Limiting Conditions for Analysis

The actual analysis for burnup credit must be performed with validated codes at limiting conditions for the SNF package. These limiting conditions apply to the generation of SNF isotopic compositions as well as the package criticality analysis. The actual values of the limiting conditions depend on the set of assemblies that they are intended to address. Hence, for most of the parameters, only the direction of the most limiting condition is addressed in this topical report.

The isotopic analysis depends on the reactor conditions during the burnup. These conditions are specific power, moderator temperature, fuel temperature, soluble boron concentration, and power versus time for the life of the fuel. The higher the specific power (MW/MTU), moderator temperature, and fuel temperature, the more reactive the fuel assembly is after a given burnup. A specific power of 60 MW/MTU bounds PWR fuel designs and does not overly burden the analysis with conservatism. The maximum core outlet moderator temperature and the maximum pellet averaged temperature should be used. The higher the average soluble boron concentration during burnup, the more reactive the fuel assembly would be following the discharge. The highest average boron concentration for any cycle for each fuel design should be used. The less time the reactor is shut down during the burnup, the more reactive the fuel assembly. Therefore, the burnup analysis should be performed as one continuous burn with no down time because this maximizes reactivity and is therefore conservative.

The criticality analysis of the SNF package must also be done at the most limiting conditions. There are three effects that are treated slightly differently for burnup credit. First, the optimum moderator density must be established for each specific package design for at least two burnup-enrichment conditions. The second consideration is the axial burnup modeling. A large database of axial burnup profiles has been developed and the most limiting shapes (as a function of burnup) have been selected. Package criticality analysis is to be performed with 18 axial nodes and the limiting shapes presented in Chapter 4. The final effect is the horizontal burnup gradient modeling. Again, a database of assembly quadrant horizontal burnup gradients has been created. Conservatively assumed gradients as a function of burnup are provided in Chapter 4. All package analysis must use these assumed burnup gradients.

7.2.4 Generation of Burnup Credit Loading Curves

Once the codes are validated and the bounding values for input to the analysis are known, burnup credit loading curves can be generated. The procedure requires determining the maximum fresh fuel enrichment and then burnup analysis of enrichments up to a limiting maximum enrichment. For each enrichment, the burnup where the SNF package design k_{eff} approximately equals, but does not exceed the USL, is determined. These values are then plotted to develop the burnup credit loading curve. The burnup plotted on the loading curve is the minimum allowable burnup, and the utility is required to reduce the burnup by the uncertainty in the burnup records.

Burnup credit loading curves should be generated for each assembly design. Separate loading curves may be generated for assemblies with removable burnable absorbers. The burnup credit loading curve will specify the minimum cooling time (up to 25 years) used in the analysis.

Cooling times longer than the minimum specified are conservative for the first 200 years of cooling.

7.2.5 Verification of Loading

The analysis of an SNF package using burnup credit results in loading criteria to identify assemblies that may be placed in a burnup credit package. These criteria provide the relationship between the minimum allowable average burnup and the initial enrichment of an assembly for a given assembly design, burnable absorber loading, and cooling time. Therefore, the package loading procedure requires knowledge of this information for a candidate assembly. This information resides in the reactor operating records. These records associate this information with a storage rack location and the ID of the assembly. Part of this record, the initial enrichment and storage rack location, is used to satisfy the criterion for current spent fuel shipments. Thus, the operational aspects of burnup credit require only an extension of the reliance of reactor records currently used for package loading. However, such an extension increases the reliance on administrative controls to ensure criticality safety. To mitigate this reliance, the burnup credit process requires the utility to verify with measurement the reactor record assembly average burnup. This verification reduces reliance on administrative controls and provides sufficient additional protection against misloading to satisfy ANSI/ANS 8.1.

7.3 RANGE OF APPLICABILITY

This topical report has a wide range of applicability for commercial power plant PWR fuel. Fifty-four chemical assays were performed which cover current commercial PWR fuel with enrichments up to 4.05 wt.% U-235, except those assemblies with integral fuel burnable absorbers. UO_2 critical experiments exist which contain configurations with a wide variety of supplemental absorbers (integral and external to the fuel assemblies), reflectors, and pin spacings. The 47 MOX critical experiments adequately cover the plutonium isotopes and Am-241. The limiting conditions (i.e., specific power, moderator and fuel temperature, ppmb, and axial burnup model) for the analysis bound the assemblies and package criticality analyses. The items that limit the range of applicability for this topical report are:

1. Burnup credit benefits can be gained from fuel burned up to 40 GWd/MTU. SNF with an assembly average burnup greater than 40 GWd/MTU shall be treated as having a burnup of 40 GWd/MTU for the purposes of this methodology.

   The highest burnup in the chemical assays was 46 GWd/MTU, but there is only one other data point above 40 GWd/MTU.

2. Enrichments above 4.05 (nominal 4) weight percent U-235 are excluded.

   Enrichment has a direct impact on criticality and an indirect impact on isotopic depletion. The criticality experiments contain enrichments up to 5.7 weight percent U-235. The chemical assays also contain a range of enrichments that can be used to establish the existence of any trend. Trends on enrichment in the isotopic concentration

prediction are not expected since it would have to be due to an error in the fission cross section, and any error that would be sufficient to cause a significant error in isotopic concentration would generally provide unacceptable errors in the criticality analysis. However, due to the limited extent of the range of enrichments exhibited in the chemical assays, the limit of 4.05 weight percent U-235 is selected.

3. Assemblies with integral fuel burnable absorbers (IFBAs) are excluded.

   No chemical assays were analyzed for fuel with IFBAs. The boron-coated IFBAs may be closely represented by the assayed pin that was next to a removable burnable absorber, but at this time it is viewed prudent to exclude such assemblies.

4. The methodology applies to SNF with cooling times ranging from 1 to 200 years.

   Cooling times less than 1 year or greater than 200 years are not of interest to current burnup credit concepts for storage or transportation.

5. MOX initial content fuel is excluded.

   No chemical assays were used from this type of fuel.

6. Reconstituted or disassembled fuel is excluded. Also excluded are fuel assemblies which have had any of their original rods removed or replaced.

   Modified or non-intact fuel assemblies may not be bounded by design basis criticality analyses.

7.4 CONSERVATISM IN THE BURNUP CREDIT METHOD

The methodology for utilizing actinide-only burnup credit described in this topical report includes substantial conservatism. The conservatisms are included to compensate for the limited knowledge of the fuel isotopic composition (including the spatial distribution), cross sections and burnup profiles, and uncertainties in the measurements and calculational tools. This section will explore some of the issues associated with the methodology's conservatisms.

Analyses have been performed to quantify the reactivity effects due to three of the conservatisms in the methodology: the bounding depletion parameters, the isotopic correction factors, and the exclusion of the fission products.[7-1] To assess each of the three effects, criticality calculations are performed using four sets with different modeling conditions. Each set consists of several combinations of typical burnups and enrichments, using a standard W17x17 assembly with a 5-year cooling time after the final cycle. The initial set represents best-estimate conditions, using nominal modeling parameters for the isotopic calculations, bias corrected isotopics, and fission products. The nominal modeling parameters represent average values for the fuel, clad and moderator temperatures, soluble boron concentration, and specific power. The bias corrected

isotopics are computed using the isotopic biases from Chapter 2, but the concentrations are not corrected for the uncertainties.

The remaining three sets vary the modeling conditions in order to be able to quantify the various effects on the system's reactivity. The second set excludes the fission products; the third set excludes fission products and uses bounding modeling parameters for the isotopic calculations. The fourth set represents the actinide-only burnup credit methodology values, which require bounding depletion parameters, use of conservative correction factors for isotopic concentrations, and no fission products. Using the various sets, the effects of each of the modeling considerations are computed at different burnups and enrichments, and are presented in Table 7-2. Results shown are differences in k_∞ between the corresponding cases.

The fission product (and U-236) conservatism shown on Table 7-2 is large. Nevertheless, since strong documentation of individual fission products' worth is not available at this time, direct credit cannot be taken for fission products. Although fission product yields can be measured, the transmutation in the reactor has little experimental verification, and thus calculated fission product concentrations cannot be easily verified. Therefore, although obviously present in SNF providing considerable negative reactivity, fission products are not directly included in the burnup credit methodology and are left as added conservatism.

Table 7-2. Conservatisms in the Actinide-Only Burnup Credit Methodology

Enrichment (wt% U-235) & Burnup (GWd/MTU)		Fission Product and U-236 Conservatism (% Δk)	Limiting Depletion Parameters Conservatism in the Mean Cask (% Δk)	Maximum Conservatism due to Statistical Treatment of the Isotopic Correction Factors (% Δk)	Estimate of the Total Conservatism in the Mean Cask (% Δk)
3.0	15	8.4	1.1	1.8	11.3
	30	13.0	3.1	2.4	18.5
	45	16.0	5.2	3.1	24.3
3.6	15	8.2	0.8	1.6	10.6
	30	12.8	2.3	2.2	17.3
	45	16.2	4.4	2.9	23.5
4.5	15	7.9	0.4	1.4	9.7
	30	12.4	1.4	1.9	15.7
	45	16.1	3.0	2.6	21.7

The other conservatisms shown on Table 7-2 are due to the modeling parameters and isotopic correction factors. Although not as large as the fission product values, considerable margin is provided by both of these bounding modeling conditions. The correction factors may not seem to be a conservatism since they are merely accounting for the uncertainty in the data. This would be logical if it was done for one isotope, but since it is done for each isotope, it implies that each isotope deviates from its expected value in the same direction (in the direction that creates more reactivity). Unfortunately, since the isotopes are all of different worths, it is not clear how to statistically combine the uncertainties. It is anticipated that future work may allow the combination of these values in a more reasonable manner.

Table 7-3 uses the same analyses results to show the change in reactivity due to burnup. The third column presents the difference in k_∞ between the zero burnup case and cases at the various burnup values for the best estimate set. The fourth column presents analogous results, but the computed difference is between the zero burnup case and the actinide-only burnup credit set. The fifth column gives the ratio of the values in columns four and three to show the reactivity percentage accounted for with actinide-only burnup credit. It is noted that when credit is taken for actinide-only burnup credit, it uses less than half of the actual reactivity change of burnup.

Table 7-3. Conservatisms in the Change in Reactivity as a Function of Burnup

Enrichment (wt% U-235)	Burnup (GWd/MTU)	Best Estimate Change in Reactivity with Burnup (% Δk)	Actinide-Only Change in Reactivity with Burnup (% Δk)	Percent of Best Estimate
3.0	15	19.4	8.0	41%
	30	34.5	16.0	46%
	45	46.6	22.2	48%
3.6	15	18.2	7.7	42%
	30	32.8	15.5	47%
	45	45.6	22.0	48%
4.5	15	16.5	6.9	42%
	30	29.9	14.1	47%
	45	42.5	20.7	49%

Tables 7-2 and 7-3 only review the conservatisms in the isotopic calculations and exclusion of the fission products and U-236. In addition to those, conservatism is also present due to using the most limiting axial burnup profiles. Again, since the profiles are possible profiles, this might not be considered a conservatism, yet most fuel assemblies have burnup profiles that do not produce

positive end effects with the actinide-only assumption. Figure 4-6 shows the population of fuel assemblies and it is obvious that the limiting profiles represent a small fraction of the assemblies. In the methodology, it is assumed that the package is full of assemblies with the limiting profile. Clearly, most packages will contain assemblies with a mix of axial profiles and hence a mean profile would be expected.

There is also the conservatism due to the horizontal burnup tilt. Although small for large packages, the effect is considerably large for four assembly packages. For this conservatism, it is not only assumed that strong horizontal gradients exist in every assembly, but that they are loaded in the most limiting way.

Other conservatisms are also introduced in the criticality validation and measurement sections. Additionally, the method does not give credit for those assemblies with reactivities below the maximum allowed. The aggregate of these below design basis reactivities provides additional criticality safety margin and conservatism.

The methodology presented in this topical report has been developed to meet the regulatory assumption of limiting k_{eff} = 0.95, which has been determined to provide an adequate safety margin. The conservatisms that have been discussed here are in excess of that margin.

7.5 TALLY OF USE OF THE FISSION PRODUCT MARGIN

At three points in this topical report (Sections 3.2, 4.1.5, and 4.2.3.3), a portion of the fission product and U-236 reactivity margin is utilized to bring closure to an issue. In each case, no real effect on k_{eff} is expected but the data to prove the assertions were either uncertain or not yet available. Table 7.4 tallies up this use of fission product and U-236 margin and compares it to the margin previously presented in Table 7.2. Note that none of the margin from limiting parameters or statistical treatment of isotopic correction factors is utilized. This is done since the fission product and U-236 margin can be stated as existing in every package where the other margins exist in the mean package. However, this additional conservatism still exists in the mean package. Each use of margin is discussed in the following paragraphs.

Due to the lack of a spent fuel critical experiment, issues have been raised that perpetuate lengthy debate. This topical report presents a strong argument that its conservative use of MOX and UO_2 critical experiments bound SNF. However, to close this issue, this topical report assigns 2% in k_{eff} of the fission product criticality margin for criticality validation concerns. The 2% value should be large enough to cover almost any concern with low enriched thermal system validation. The literature has been reviewed for the maximum deviations from the mean observed for low enriched thermal lattice experiments. No deviations greater than 2% have been observed for any code system (see Chapter 3, Section 3.2). The critical experiments analyzed cover all the expected cask features and all the isotopes for which credit is sought. Although the particular combination of isotopes and cask features are not contained in the experimental database, the use of a conservative method with a reserve margin of the maximum deviation from all experiments should resolve all concerns.

Table 7-4. Tally of the Use of the Fission Product Margin

Enrichment (wt% U-235) & Burnup (GWd/MTU)		Fission Product and U-236 Conservatism (% Δk)	Criticality Validation Issues (% Δk)	Effect if Control Rods Were Inserted During Depletion (% Δk)	Computer Code Adequacy Issues (% Δk)	Remaining Margin (% Δk)
3.0	15	8.4	2	3.3	1	2.1
	30	13.0	2	3.3[1]	1	6.7
	45	16.0	2	3.3[1]	1	9.7
3.6	15	8.2	2	2.1	1	3.1
	30	12.8	2	2.1[1]	1	7.7
	45	16.2	2	2.1[1]	1	11.1
4.5	15	7.9	2	1.0	1	3.9
	30	12.4	2	1.0[1]	1	8.4
	45	16.1	2	1.0[1]	1	12.1

[1]Control rod effect from more than 15 Gwd/MTU of rodded burnup is not considered credible

The control rods in a PWR are generally out during power operation. However, it is allowable to have the lead bank inserted in a fraction of the core. Fuel shuffling between cycles reduces the fraction of fuel that may be impacted by control rods for more than one cycle. Full insertion for the equivalent of 15 GWd/MTU, about one cycle, is considered to be an upper bound for those assemblies impacted by control rod insertion. The lead control bank covers less than 7% of the fuel assembly locations. Since a package will have multiple assemblies, whatever effect that might exist due to control rod insertion would be diluted. The control rod effect assumes the full length of the assembly was covered by control rods. This assumption leads to at least a factor of two over prediction. Since this effect applies to a very small fraction of the assemblies and it can be covered by the fission product margin, there is no need to require utilities utilizing a small cask to maintain the reactor records for each assembly as to the control rod positions, and to penalize those assemblies that are impacted by control rods.

Adequacy of the computer code packages has been discussed at length. For example, it is known that the computer code used may not always converge to the correct answer when there are dual flux peaks unless the user explicitly assures sufficient neutron sampling in all regions. To answer convergence adequacy concerns, it is now required that the cask vendor check that each of the 18 nodes are sufficiently sampled. To complete closure of computer code adequacy issues, 1% in k_{eff} of the fission product and U-236 is assigned. One percent was selected since it is greater than one standard deviation seen in international benchmarks.

In summary, note that the remaining margin shown in Table 7-4 for greater than 15 GWd/MTU is more than half of the fission product margin. At 15 GWd/MTU, at least 25% of the fission product margin remains. Experiments performed to determine fission product worth have been performed in Europe. Neither the analysis done in France nor United Kingdom showed any prediction of isotopic worths with errors greater than 25%.[7-2, 7-3] Therefore, fission product margin adequately covers the conservative bounds of the impacts of these issues.

7.6 SUMMARY OF NRC APPROVALS SOUGHT

This topical report seeks NRC concurrence that: 1) the data presented are sufficient to validate actinide-only burnup credit, and 2) the method presented to provide a basis for using burnup credit is valid. This topical report is specifically seeking NRC acceptance of the following:

1. The PWR fuel post irradiation examination assay data selected for isotopic inventory bias and uncertainty determination is sufficient for validating the selected actinide composition in spent fuel.

2. The statistical procedure proposed for establishing isotope-specific biases and correction factors is a conservative method to account for isotopic concentration changes during burnup.

3. The selection of the 47 MOX criticality experiments for actinide-only burnup credit analysis.

4. The use of the most limiting USL from the UO_2 and MOX criticality experiment analysis.

5. Confirmations that the methods for criticality validation established in NUREG/CR-5661 apply for burnup credit with the supplement of the MOX critical experiments as specified in points 3, 4, and 5.

6. A single cycle burnup at a specific power of 60 MW/MTU conservatively bounds the effects of specific power and operating history on isotopic concentrations.

7. The use of the maximum cycle average dissolved boron concentration conservatively accounts for soluble boron effects on isotopic concentrations.

8. The reactivity of the spent fuel is maximized by setting the fuel temperature to the maximum pellet averaged temperature.

9. The use of the maximum core outlet temperature in determining the moderator density for depletion produces conservative isotopic concentrations.

10. The method presented for determining optimum moderation in the SNF package is acceptable.

11. The use of the selected limiting axial burnup profiles for burnup credit conservatively captures the end effects.

12. The selected horizontal gradients and use of the most limiting arrangement in the package analysis sufficiently model horizontal burnup effects.

13. The method used to generate two-parameter loading curves (i.e., burnup and initial enrichment) for specifying package burnup credit loading requirements is suitable and appropriate.

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APPENDIX A

ACRONYMS

AEG	Average Energy Group causing Fission
ALA	Average Lethargy for Absorption
ALC	Average Lethargy for Capture
ALF	Average Lethargy for Fission
ANS	American Nuclear Society
ANSI	American National Standards Institute
ATM	Approved Testing Materials
BAR	Burnable Absorber Rod
BCL	Battle Columbus Laboratory
BUC	Burnup Credit
CE	Combustion Engineering
CFR	Code of Federal Regulations
CRWMS	Civilian Radioactive Waste Management System
CSAS	Criticality Safety Analysis Sequence
DOE	Department of Energy
EPRI	Electric Power Research Institute
GWd/MTU	Gigawatt Day Per Metric Tons Uranium
HEDL	Hanford Engineering Development Laboratory
HLW	High-Level Radioactive Waste
IFBA	Integral Fuel Burnable Absorber
LPB	Lower Prediction Band
LWR	Light Water Reactor
M&O	Management and Operating Contractor
MCC	Materials Characterization Center
MOX	Mixed Oxide
MPC	Multi-Purpose Canister
MW/MTU	MegaWatt Per Metric Tons Uranium
NRC	Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act of 1982, as amended
OCRWM	Office of Civilian Radioactive Waste Management

ORNL	Oak Ridge National Laboratory
PNL	Pacific Northwest Laboratories
ppmb	Parts per Million Boron
PUP	Plutonium Utilization Program
PWR	Pressurized Water Reactor
QA	Quality Assurance
QARD	Quality Assurance Requirements and Description
RG	Regulatory Guide
RSIC	Radiation Shielding Information Center
SAR	Safety Analysis Report
SCALE	Standardized Computer Analyses for Licensing Evaluation
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratories
USL	Upper Safety Limit

APPENDIX B

GLOSSARY

27BURNUPLIB - The SCALE 4.2 27 group burnup library containing ENDF/B-IV (actinides) and ENDF/B-V (fission products) neutron cross section data. The cross sections are used in SAS2H fuel depletion and CSAS25 criticality analysis sequence calculations.

Absorber - A neutron-capture material. Absorber nuclides have a large neutron absorption cross section relative to their fission cross section.

Actinide-Only Burnup Credit - Credit for the reactivity change from fresh fuel to spent fuel accounting only for the change in actinide isotopic concentrations. Credit for the addition of fission product absorbers is not taken.

Actinides - A chemical group which contains, for the purpose of this report, U, Np, Pu, Am, and Cm.

Areas of Applicability - The ranges of material compositions and geometric arrangements within which the bias of a calculational method is established.

Assembly Identifier - A unique string of alphanumeric characters which identify an assembly, bundle, or canister from a specific reactor in which it has been irradiated. Must be consistent with other submissions to the DOE/NRC; that is, Annex B, previous Form RW-859, and DOE/NRC Form 741.

Axial Burnup Distribution - The variability in SNF burnup along the length of an assembly. Typically, burnup is highest in the center region and lowest at the ends.

Basket - The internal component of a spent fuel storage, transportation, or disposal package that provides structural support for individual spent fuel assemblies and assures a subcritical geometry. The basket also functions to provide thermal conductivity to remove spent fuel decay heat.

Benchmark (noun) - A well-specified experiment that can be used to validate analytical methods. Accurate descriptions of the experimental configurations and materials are provided along with method descriptions and detailed results (including uncertainties and tolerances).

Benchmark (verb) - Verification of the area(s) of applicability and bounds of an analysis method by comparison to either experimental results or the results of another analysis method that has been verified experimentally.

Bias - A measure of the systematic disagreement between the results calculated by a method and experimental data. The uncertainty in the bias is a measure of both the precision of the calculation and the accuracy of the experimental data.

BONAMI-S - A SCALE 4.2 module that performs resonance self-shielding calculations for isotopes that have Bondarenko data associated with their cross sections. The module is called by the SAS2H fuel depletion and CSAS25 criticality analysis sequences.

Burnable Absorbers - Absorbers placed in selected locations in a reactor core, external to the fuel rods, to enhance reactivity and power distribution control. Burnable absorbers are manufactured from materials that include a neutron absorber, which is converted to a nuclide with low absorption cross section as a result of neutron absorption. Similar reactor core reactivity control benefits are achieved with integral fuel burnable absorbers, which are added to the fuel matrix during fuel manufacture.

Burnup - 1) the process of fuel being consumed by fissioning; 2) a measure of the amount of energy obtained from fuel as the fuel fissions, which is expressed as the amount of energy produced per unit of fuel weight or the percentage of fissile atoms consumed during irradiation.

Burnup Credit (BUC) - The process of accounting for the operating history of spent nuclear fuel in criticality safety calculations and fuel loading operating procedures and controls.

Burnup Credit Isotopes - The isotopes selected to represent the composition of spent fuel in the burnup credit method.

Burnup Credit Loading Curve - A line plotted on an X-Y graph through limiting combinations of fuel assembly initial enrichment and required minimum burnup established using the burnup credit method. The curve specifies the criticality control design criteria and serves as the operational limit for selecting fuel assemblies for loading into a burnup credit SNF package.

Burnup Credit Method - The mathematical equations, approximations, assumptions, associated numerical parameters (e.g., cross sections), and calculational procedures that yield the burnup credit loading curve.

Burnup Credit Package - A storage, transportation, or disposal package designed to incorporate the operating history of spent nuclear fuel in criticality safety calculations and fuel loading operating procedures and controls.

Candidate Assembly - A spent fuel assembly determined by procedure to meet minimum burnup and any other requirements specified by a burnup credit SNF package Certificate of Compliance and the supporting Safety Analysis Report.

Cooling Time - The time since a spent fuel assembly is permanently discharged from the operating reactor.

Critical - A nuclear system is critical when the total number of fission neutrons produced during a time interval is equal to the total number of neutrons lost by absorption and leakage during the same interval (i.e., $k_{eff} = 1$).

CSAS - SCALE 4.2 sequences that perform criticality analysis calculations. CSAS analysis sequences are standardized automated procedures that process SCALE 4.2 cross sections using BONAMI-S and NITAWL-S, and perform a criticality analysis using KENO V.a.

Depletion - Isotopic transmutations occurring while the fuel is in the reactor core and producing power.

Double Contingency Principle - As adapted from ANSI/ANS-8.1, criticality control systems and procedures should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in conditions or failures in procedural controls before a criticality accident is possible.

End Effect - The k_{eff} difference between an axially burnup-dependent criticality calculation and an axially uniform criticality calculation.

Enrichment - A measure of the atom or weight percent of a particular isotope when it is increased above its abundance as found in nature.

Fissile Isotope - An isotope that is capable of fissioning when bombarded by a thermal neutron.

Fission Products - The bi-product nuclei resulting from fission events.

Fresh Fuel - Nuclear fuel that has not been exposed to any significant neutron sources.

Fresh Fuel Assumption - A term used to describe the historic method of modeling fuel for criticality analysis where it is assumed that the fuel is at its initial enrichment.

H/U - The ratio of hydrogen to uranium in a system containing uranium fuel and hydrogeneous moderator.

Independent Burnup Verification - An accurate, relative indication of spent fuel assembly burnup correlated from neutron and gamma emission measurements and reactor records for assembly initial enrichment and cooling time since final discharge from the reactor.

Intact Fuel Assemblies - "As-received" by a reactor operator, in those characteristics important to the criticality safety analysis; i.e., all original fuel pins are present and assembly array characteristics, including pin pitch, and guide and instrument tube characteristics are unaltered from the original as-manufactured design configuration. The presence of irradiated burnable absorber rods in the guide tube locations is specifically identified as an "intact" assembly. Intact Fuel Assemblies are potential candidates for loading into a burnup credit package. The presence of fuel pins in guide or instrument tube locations is specifically identified as "**not** intact" and **not** acceptable for loading in a burnup credit package.

Integral Fuel Burnable Absorbers - Burnable absorbers integral to the fuel pin. These include Gd or Er mixed in the pellet or a boron compound coating on the pellet.

k_{eff}, Effective Multiplication Factor - The ratio of the neutron production rate by fission in a nuclear system to the rate of neutron loss by leakage and absorption

k_∞, Infinite Multiplication Factor - The ratio of the neutron production rate from fission in an infinite nuclear system to the rate of neutron loss by absorption (i.e., leakage is zero).

KENO V.a - A SCALE 4.2 module that performs a 3-D multigroup Monte Carlo criticality analysis. The module is called by the CSAS25 criticality analysis sequence.

Loading Criteria - Fuel loading requirements, limits, and controls specified by a burnup credit SNF package Certificate of Compliance and the supporting Safety Analysis Report.

Modeling Parameters - Material and geometric characteristics of a system necessary to describe the system for calculational purposes, which, when varied, influence the margin of subcriticality.

Moderator - Material incorporated into a nuclear system to slow neutrons to lower energy levels by collision processes.

Neutron Cross Section - A proportionality constant describing the extent to which neutrons interact with nuclei of a material.

NITAWL-S - A SCALE 4.2 module that applies a Nordhiem resonance self-shielding correction to isotopes having resonance parameters. The module is called by the SAS2H fuel depletion and CSAS25 criticality analysis sequences.

Non-specification Assembly - A spent fuel assembly determined by procedure to not meet minimum burnup or other requirements specified by a burnup credit SNF package Certificate of Compliance and the supporting Safety Analysis Report.

ORIGEN-S - A SCALE 4.2 module that performs both isotope generation and depletion calculations for a specified reactor fuel history. ORIGEN-S is called by the SAS2H analysis sequence.

Package - The shielded container together with its radioactive contents as prepared for storage, transport, or disposal.

Package Capacity - The number of individual spent fuel assemblies that can be physically inserted into a particular transportation package.

Qualified Assembly - A spent fuel assembly determined by procedure to meet minimum burnup requirements specified by a burnup credit SNF package Certificate of Compliance and the supporting Safety Analysis Report, and verified by measurement to exhibit characteristics consistent with reactor records with regard to initial enrichment, burnup, and cooling time.

Reactivity - A measure of the departure of a nuclear system from critical.

Reactor Records - Utilization facility records pertaining to spent nuclear fuel manufacture, irradiation history, and current storage location.

SAS2H - A SCALE 4.2 sequence that performs fuel isotope generation and depletion analysis calculations and analysis of spent fuel packages. The SAS2H analysis sequence is a standardized automated procedure which processes SCALE 4.2 cross sections using BONAMI-S, NITAWL-S, XSDRNPM-S and COUPLE, and performs a fuel nuclide generation, depletion and decay analysis using ORIGEN-S.

SCALE 4.2 - A modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols, I, II, and III. Available from Radiation Shielding Information Center, Oak Ridge National Laboratory, as CCC-545.

Special Nuclear Material - 1) Plutonium, uranium 233, uranium enriched in isotope 233 or isotope 235, and any other material determined as special nuclear material pursuant to Section 51 of the Atomic Energy Act, but does not contain source material or 2) any material artificially enriched by any of the foregoing, but does not include source material.

Specific Power (MW/MTU) - The amount of power produced per metric ton of uranium originally in the fuel.

Spent Nuclear Fuel - Burned fuel that has been permanently withdrawn from a nuclear reactor.

Spent Nuclear Fuel Package - This is a general term to encompass transportation casks, storage containers, waste packages, or a multi-purpose canister.

Staged Fuel Assembly - A qualified fuel assembly that is physically positioned in preparation for SNF package loading consistent with the package loading procedure.

Subcritical - A nuclear system is subcritical when the total number of fission neutrons produced during a time interval is less than the total number of neutrons lost by absorption and leakage during the same interval (i.e., $k_{eff} < 1$).

Thermal Neutrons - Neutrons that are in substantial thermal equilibrium with the core material and are the primary means for inducing fission in fissile material.

Upper Safety Limit (USL) - The highest value of k_{eff} allowed so that subcriticality is ensured. This limit accounts for all the biases, uncertainties, administrative margins, and licensing assumptions.

Validation - A process to demonstrate that analytical methods meet predetermined requirements.