PG&E Response to NRC Request for Additional Information Regarding License Amendment Request 01-02, "Credit For Soluble Boron In The Spent Fuel Pool Criticality Analysis "

Question 1

The cover letter (DCL-01-096) states that the criticality analysis was performed using a methodology analogous to that of WCAP-14416-NP-A. Methodologies that deviate from approved methodologies must be submitted, reviewed, and approved for a particular application. Furthermore, portions of WCAP-14416-NP-A can no longer be relied upon as an approved methodology by the NRC staff for licensing actions. The known nonconservatisms in axial burnup biases as applied in WCAP-14416-NP-A were consolidated by Westinghouse in Letter NSAL-00-0015, "Axial Burnup Shape Reactivity Bias," dated December 6, 2000. By letter dated August, 2001 NRC informed Westinghouse that portions of WCAP-14416-NP-A could no longer be referred to in licensing actions. The analysis in your present submittal included an axial burnup distribution represented by a four zone axial model to account for the axial burnup bias non-conservatisms of WCAP-14416-NP-A. This model was stated to be adequate to represent a fuel assembly. References 14 and 15 (of the Westinghouse Spent Fuel Pool Criticality Report A-DP1-FE-0001, Enclosure 5 to DCL-01-096) were identified as the bases to support your position. However, these references, namely DOE/RW-0472 and A-GEN-FE-0118 have not been reviewed and approved by NRC staff. Submit the referenced reports for review.

PG&E Response to Question 1

The Westinghouse Electric Company LLC (Westinghouse) Spent Fuel Pool Criticality Report A-DP1-FE-0001, contained in Enclosure 5 to PG&E Letter DCL-01-096, "License Amendment Request 01-02, Credit For Soluble Boron In The Spent Fuel Pool Criticality Analysis," dated September 13, 2001, referenced U.S. Department of Energy Topical Report DOE/RW-0472, Revision 1, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages," dated May 1, 1997, for the limiting axial burnup profile data. The burnup profile contained in DOE/RW-0472, Revision 1, is based on a database of 3169 axial-burnup profiles for pressurized water reactor fuel assemblies compiled by Yankee Atomic Company. DOE/RW-0472, Revision 1 is contained in Enclosure 2 for NRC review. Report DOE/RW-0472, Revision 1, was originally submitted to Dr. William Kane, Director of Spent Fuel Project Office, U.S. NRC by Mr. Christopher A. Kouts, Director of Storage and Engineering Technical Division, Department of Energy Office of Civilian Radioactive Waste Management, by a Letter dated May 15, 1997.

The Westinghouse Spent Fuel Pool Criticality Report A-DP1-FE-0001 referenced Westinghouse Report A-GEN-FE-0118, Revision 0, "Isotopic Number Densities for Discharged Westinghouse 17x17 Fuel Assemblies," dated November 15, 2000, for the adequacy of a four-zone axial model to represent the spent fuel assembly. Proprietary

and non-proprietary versions of Westinghouse Report A-GEN-FE-0118, Revision 0, are contained in Enclosure 3 for NRC review.

Question 2

NUREG/CR-6683 entitled, "A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage," published September 2000, demonstrates that reactivity equivalencing results in significant under-estimation of K_∞ in soluble boron calculations. In light of the results provided in NUREG/CR-6683, the staff no longer considers reactivity equivalencing as appropriate for calculating boron credit since the errors can be greater than the allowable credit for boron. The errors, if unaccounted for, could allow criticality. The use of reactivity equivalencing cannot support a request for soluble boron credit as presently submitted. Provide calculations which do not rely upon reactivity equivalencing for boron credit or define an approach to address the staff's concerns.

PG&E Response to Question 2

NUREG/CR-6683 questions the use of the reactivity equivalency method which is defined as a method to equate an array of fresh fuel assemblies and their enrichments, that have been shown to be acceptable for storage, into an array of irradiated assemblies with different initial enrichments, decay times, and burnup absorber concentrations. The Westinghouse spent fuel pool criticality analysis contained in Report A-DP1-FE-0001, Revision 0, did not use the reactivity equivalency method as described in NUREG/CR-6683.

Section 3.6.1 of Report A-DP1-FE-0001, Revision 0, discusses the method used to determine the soluble boron concentration required to maintain K_{eff} less than 0.95. Table 3.6-1 contains the data used to model a repeating 2x2 array of one fresh (no burnup) fuel assembly checkerboarded with three burned fuel assemblies. The description section of Table 3.6-1 states that the fresh assembly modeled had an enrichment of 4.90 weight percent (w/o) Uranium 235 (^{235}U) and the burned assemblies modeled had an initial enrichment of 5.00 w/o ^{235}U and a burnup of 55 gigawatt days per metric ton of uranium (GWD/MTU). The burned assembly was modeled using depleted fuel isotopics with the effect of all fission products and actinide absorbers directly included.

Section 3.6.2 of Report A-DP1-FE-0001, Revision 0, discusses the method used to convert the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel assembly burnup values to a soluble boron concentration necessary to compensate for these two uncertainties. Although this method for conversion of the fuel uncertainties to an equivalent boron concentration was called "reactivity equivalencing" in Report A-DP1-FE-0001, it is not the same as the reactivity equivalencing method as described in NUREG/CR-6683. The reactivity equivalencing method as described in NUREG/CR-6683 concerns equating fresh fuel assemblies, without the fission products

and actinide absorbers, and their enrichments to burned assemblies with different initial enrichments. It is noted that the uncertainty in fuel assembly reactivity and fuel assembly burnup were equated to a soluble boron concentration with the data contained in Table 3.6-1 of Report A-DP1-FE-0001, Revision 0, which was generated by modeling the burned fuel assemblies with depleted isotopics.

Section 3.6.3 describes the method used to determine the soluble boron concentration required to maintain K_{eff} less than or equal to 0.95 under accident conditions. Table 3.6-3 contains the data used to model the worst case fuel mishandling accident, which is the misplacement of a fresh fuel assembly adjacent to another fresh fuel assembly. The description section of Table 3.6-3 states that the fresh assembly modeled for the limiting fuel mishandling accident had an enrichment of 5.00 w/o ^{235}U and the burned assembly modeled had an initial enrichment of 5.00 w/o ^{235}U and a burnup of 55 GWD/MTU. The accident scenario simulated the burned fuel assemblies by modeling the effect of the fission products and depleted isotopics in the fuel assembly. Therefore, the soluble boron concentration determined to maintain K_{eff} less than 0.95 under the worst accident condition was determined based upon a realistic representation of the burned fuel assemblies and not a reactivity equivalency method.

In conclusion, the spent fuel pool criticality analysis specifically modeled burned assemblies, with the effect of fission products and depleted isotopics directly included. The method used in this analysis, to determine the soluble boron concentration required to maintain K_{eff} less than 0.95 and to determine the soluble boron concentration required to maintain K_{eff} less than or equal to 0.95 under accident conditions, specifically modeled burned fuel assemblies and did not credit reactivity equivalent fuel assemblies. Therefore, the spent fuel pool criticality analysis did not use the reactivity equivalency method as described in NUREG/CR-6683.

Enclosure 2
PG&E Letter DCL-02-022

U.S. Department of Energy Topical Report DOE/RW-0472 Revision 1 "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages"

DOE/RW- 0472 Rev. 1

Office of Civilian Radioactive Waste Management

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

Revision 1

May 1997

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Washington, DC 20585

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-236, 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 spent nuclear fuel package. Fifty-seven UO_2, UO_2/Gd_2O_3, and UO_2/PuO_2 critical experiments have been selected to cover anticipated conditions of SNF. 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 NRC licensed utility's burnup record is compared to this minimum burnup requirement after the utility accounts for the uncertainty in its record. 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 and that measurement must be within 10% of the utility burnup record for the assembly to be accepted. The measurement device must be accurate to within 10%.

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."

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.) 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.

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. 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 burned up to 50 GWd/MTU. SNF with an assembly average burnup greater than 50 GWd/MTU shall be treated as having a burnup of 50 GWd/MTU for the purposes of this methodology.
- Enrichments above 5 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 100 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 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 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 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 constant measurements of reactivity and power distributions at power plants. Source terms generated for thermal and shielding analyses have also shown good agreement 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 was used. This demonstration resulted in validation of this computer code system for actinide-only burnup credit.

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

- That 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

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

- That the SAS2H sequence of the SCALE 4.2 code system using 27BURNUPLIB cross sections has been validated and that appropriate isotopic correction factors have been determined.

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:

- That the 57 criticality experiments selected are sufficient for validating computer codes for actinide-only burnup credit analysis

- That trending analyses on the effect on k_{eff} due to variations in spectra, initial enrichment, pellet outside diameter, and the soluble boron concentration are adequate

- That the method of determining the upper safety limit is adequate

- That the use of the developed USL with SCALE 4.2 code system with the 27BURNUPLIB and with a 0.05 Δk_m administrative safety margin is acceptable to perform actinide-only burnup credit criticality safety calculations in SNF package design.

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 spent nuclear fuel 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:

- That 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

- That the use of the maximum cycle average dissolved boron concentration conservatively accounts for soluble boron effects on isotopic concentrations

- That the reactivity of the spent fuel is maximized by setting the fuel temperature to the maximum pellet averaged temperature

- That the use of the maximum core outlet temperature in determining the moderator density for depletion produces conservative isotopic concentrations

- That the method presented for determining optimum moderation in the SNF package is adequate

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

- That 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 necessary due to variations in fuel assembly designs (i.e., Westinghouse versus BW designs). In addition, separate curves can be generated for fuel that was burned with removable burnable absorbers. 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 loading of spent nuclear fuel 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. This topical report presents a generic loading procedure that enhances the administrative controls with a physical measurement to verify the reactor records for loading burnup credit packages.

This burnup verification includes measurements of neutron and/or gamma emissions from spent fuel assemblies using any measurement system which meets specific guidelines. These measurements are correlated to the SNF assembly burnup data obtained from the reactor records. Any anomaly in the declared burnup, initial enrichment, and cooling time of the assembly would be detected in this burnup verification technique.

This topical report is seeking NRC acceptance of the burnup verification procedure and guidelines.

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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.

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.) 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.

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. 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. 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.

When reviewed and accepted by the NRC, this topical report will serve as a criterion document for criticality control analysts and 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.

There are three general areas where the DOE is requesting NRC acceptance of the actinide-only burnup credit methodology. First, the data presented are 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 the 57 critical experiments used to validate the burnup credit criticality calculations. Second, the burnup credit methodology presented is acceptable. This includes the analytical techniques and the burnup credit loading procedures. Third, that the SCALE-4.2[1-4] computer code package utilizing the 27BURNUPLIB has been validated and is acceptable for performing burnup credit criticality analyses. 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. It also provides related verification requirements for loading SNF into a transportation package that has been licensed for burnup credit. 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 not included in the scope of this report. The DOE plans to submit another topical report in the future to address the additional negative reactivity effect from the buildup of fission product neutron absorbers.

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 50 GWd/MTU. SNF with an assembly average burnup greater than 50 GWd/MTU shall be treated as having a burnup of 50 GWd/MTU for the purposes of this methodology.
- Enrichments above 5 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 100 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 includes limits on the range of applicability that also include a spectral index and specific power. Both of the spectral index and specific power have a range that covers all commercial SNF. Criticality validation covers all current SNF packages, however, each package vendor should confirm that its design is covered by the features in the criticality experiments. For example, hafnium absorbers are not included in the current set of criticals recommended in this report.

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 of SNF are established in 10 CFR Part 71. 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) and § 71.61.

The burnup credit methodology presented in this topical report is consistent with the general design criteria specified in 10 CFR Parts 71. 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 distinguishes 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 was used. This demonstration resulted in validation of this computer code system for actinide-only burnup credit.

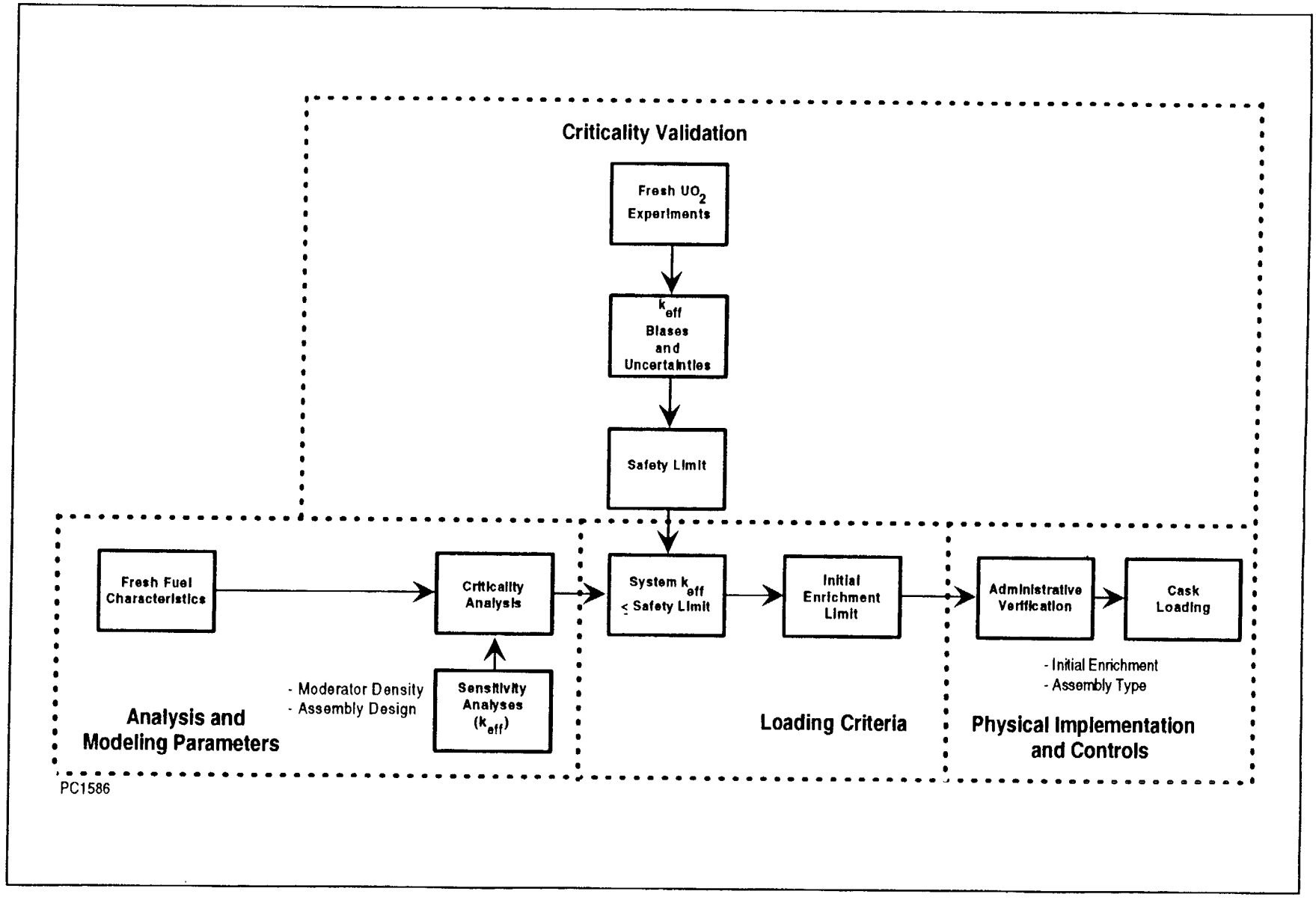

Figure 1-1. Fresh Fuel Assumption Procedure

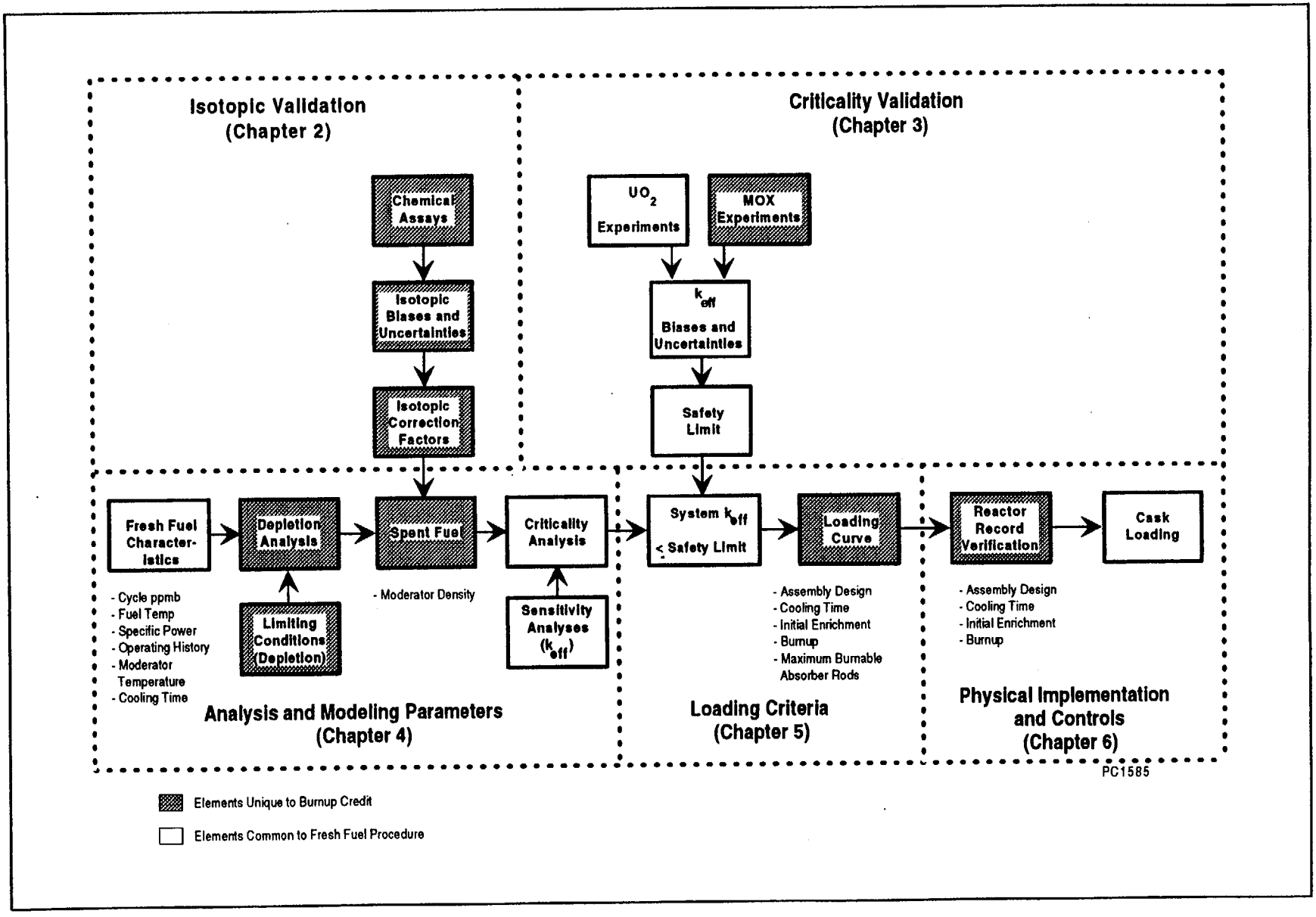

Figure 1-2. Burnup Credit Procedure

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

- That 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

- That the statistical procedure proposed for establishing isotope-specific biases and correction factors result in a conservative method to account for isotopic concentration changes during burnup

- That the SAS2H sequence of the SCALE 4.2 code system using 27BURNUPLIB cross sections has been validated and that appropriate isotopic correction factors have been determined.

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 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:

- That the 57 criticality experiments selected are sufficient for validating computer codes for actinide-only burnup credit analysis

- That trending analyses on the effect on k_{eff} due to variations in spectra, initial enrichment, pellet outside diameter, and the soluble boron concentration are adequate

- That the method of determining the upper safety limit is adequate

- That the use of the developed USL with SCALE 4.2 code system with the 27BURNUPLIB and with a 0.05 Δk_m administrative safety margin is acceptable to perform actinide-only burnup credit criticality safety calculations in SNF package design.

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 spent nuclear fuel 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:

- That 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

- That the use of the maximum cycle average dissolved boron concentration conservatively accounts for soluble boron effects on isotopic concentrations

- That the reactivity of the spent fuel is maximized by setting the fuel temperature to the maximum pellet averaged temperature

- That the use of the maximum core outlet temperature in determining the moderator density for depletion produces conservative isotopic concentrations

- That the method presented for identifying and demonstrating optimum moderation in the SNF package is adequate

- That the use of the selected limiting axial burnup profiles for burnup credit conservatively captures the end effects

- That 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 BW designs). Also, separate curves may be generated for fuel that was burned 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.

1.6.5 Physical Implementation and Controls

The loading of spent nuclear fuel 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." The previous sections introduced the methodology for determining a conservative reactivity for the SNF assemblies. In addition, this topical report presents a generic loading procedure that enhances the administrative controls with a physical measurement to verify the reactor records for loading burnup credit packages.

This burnup verification includes measurements of neutron and gamma emissions from spent fuel assemblies using any measurement system which meets specific guidelines. These measurements are correlated to the SNF assembly burnup data obtained from the reactor records. Any anomaly in the declared burnup, initial enrichment, and cooling time of the assembly would be detected in this burnup verification technique.

This topical report is seeking NRC acceptance of the burnup verification procedure and guidelines described in Chapter 6.

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 presents 57 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}. The technique seeks a correlation of the data to parameters that influence k_{eff}. The parameters used are a spectral index, initial enrichment, pellet outside diameter, and the soluble boron concentration. The CSAS sequences of the SCALE 4.2 code package with 27BURNUPLIB is used to demonstrate this process.

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 illustrates the steps and procedures for selecting SNF for loading into a burnup credit package. The use of reactor records to qualify spent fuel assemblies for loading into a burnup credit package is discussed. Guidelines are established for measurement systems to be used for verification of reactor record burnup values, and examples of measurement systems which could potentially meet the guidelines are described.

Chapter 7 summarizes the topical report and reviews the steps for implementing burnup credit.

Chapter 8 provides bibliographic information for references.

Appendix A demonstrates the application of the burnup credit methodology by providing sample calculations performed by cask vendors using the methodology presented in this report.

Appendix B provides descriptions of burnup measurement systems provided by the measurement system vendors.

Appendix C contains an acronym list, and Appendix D 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. Np-237 is not considered at this time due to large deviations between the calculated and measured values. Therefore, the selected actinide isotopes to be included in the burnup credit criticality analysis methodology are U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241.

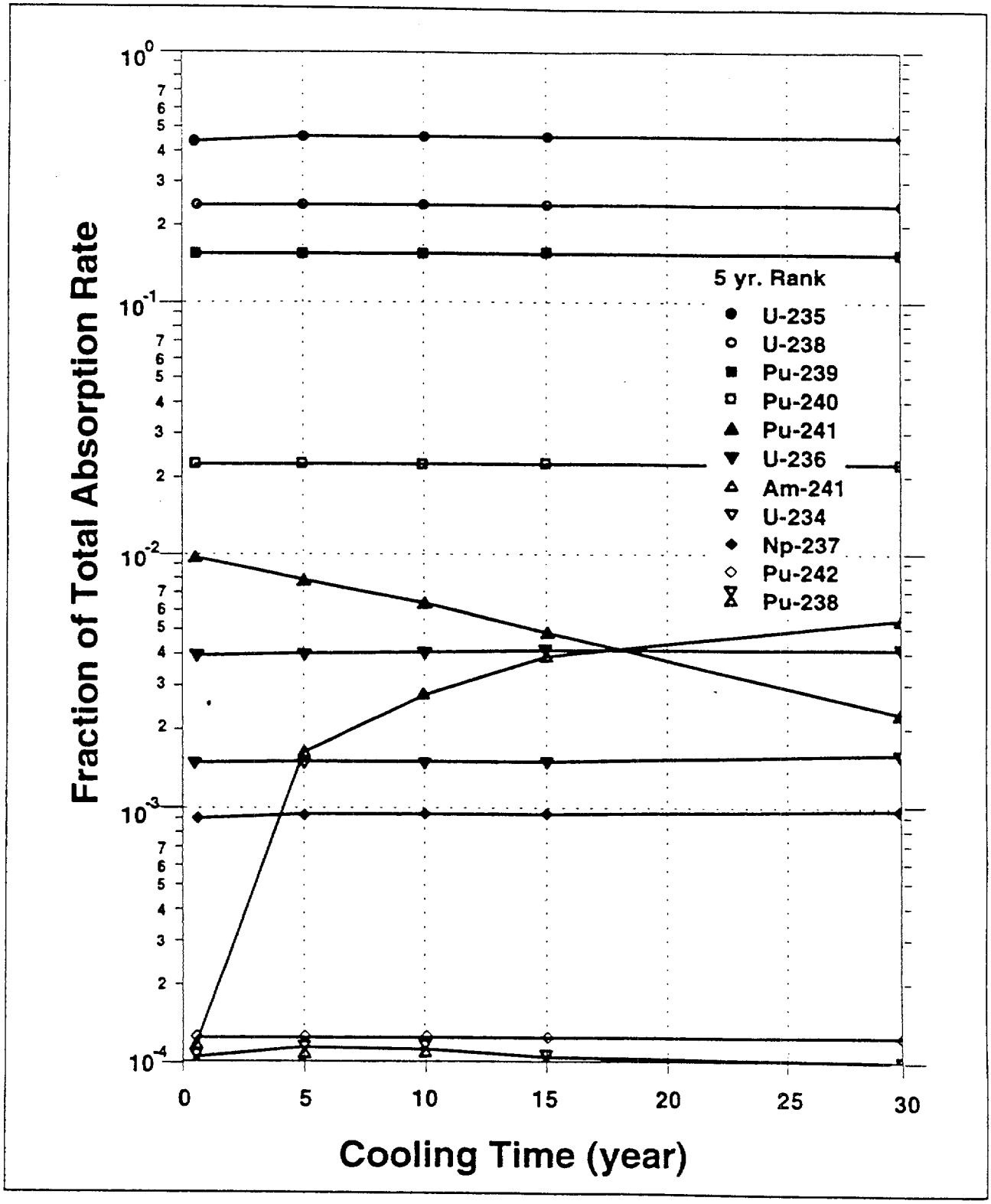


Figure 2-1. Fractions of Neutrons Absorbed by Major Actinides at Various Cooling Times, 3.6 wt. % U-235, 10 GWd/MTU

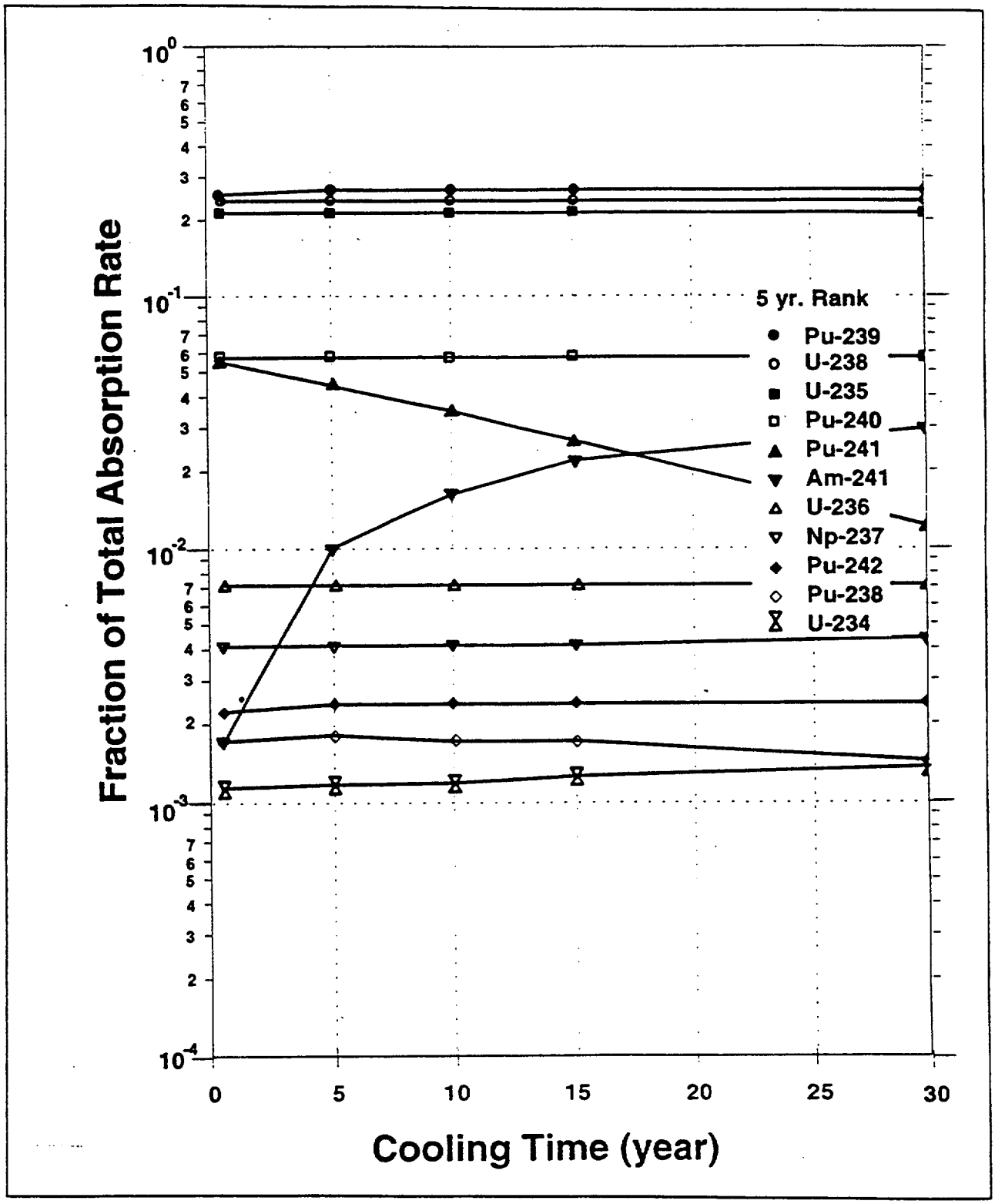


Figure 2-2. Fractions of Neutrons Absorbed by Major Actinides at Various Cooling Times, 3.6 wt. % U-235, 30 GWd/MTU

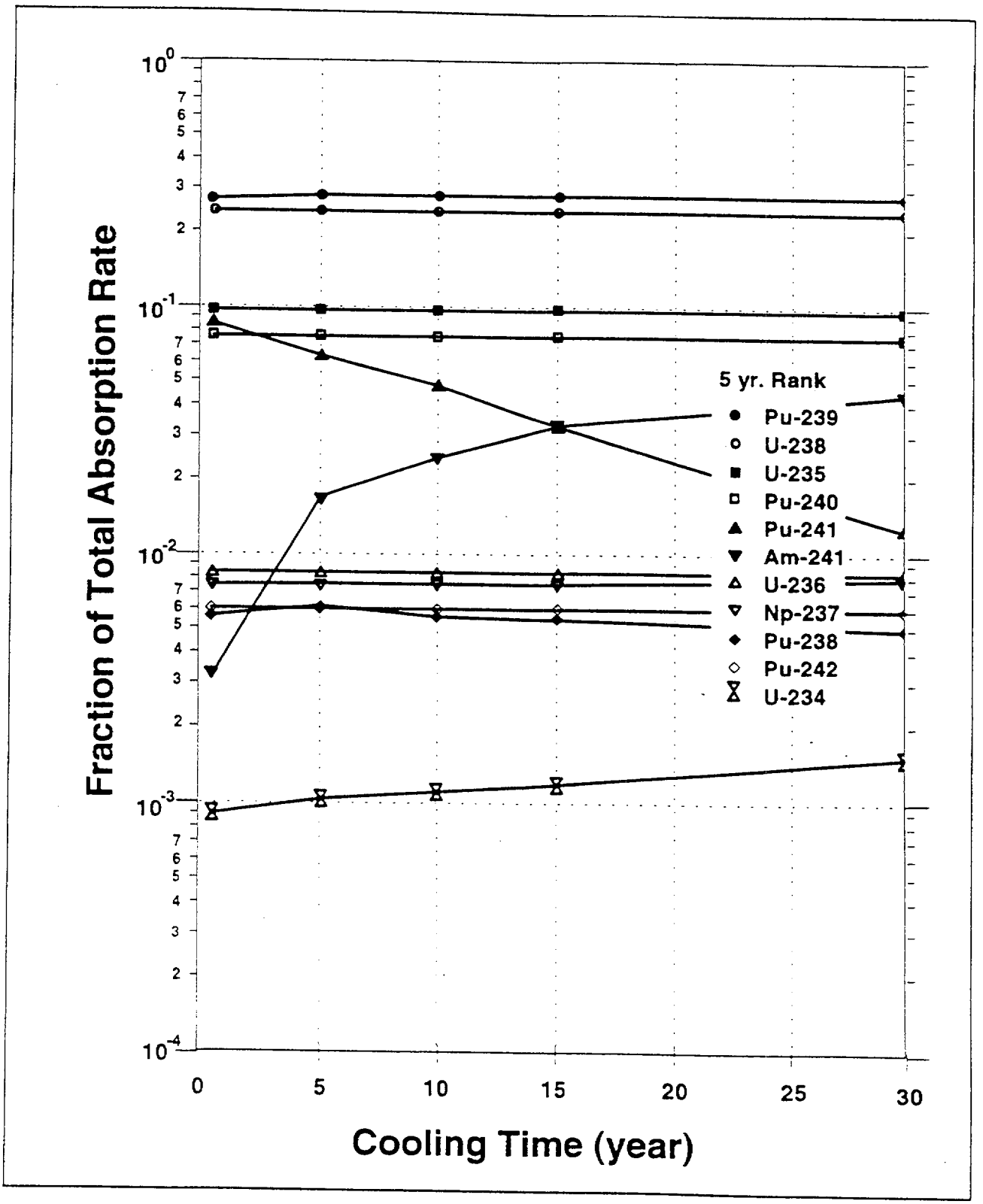


Figure 2-3. Fractions of Neutrons Absorbed by Major Actinides at Various Cooling Times, 3.6 wt. % U-235, 50 GWd/MTU

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 assays data.

The fuel assemblies analyzed at the MCC, which was 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 include 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 are 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. Therefore, 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 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 are 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 are 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 are 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 nine 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 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 2-1. 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.

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 100	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.

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

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

$$dN^{U236}/dt = (N^{U235}\sigma_c{}^{U235} - N^{U236}\sigma_a{}^{U236})\phi$$

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

$$dN^{Pu238}/dt = \lambda^{Np238}N^{Np238} - N^{pu238}\sigma_a{}^{Pu238}\phi \qquad \text{See Note 1}$$

$$dN^{Pu239}/dt = (N^{U238}\sigma_c{}^{U238} + N^{Pu238}\sigma_c{}^{Pu238} - N^{pu239}\sigma_a{}^{Pu239})\phi \qquad \text{See Note 2}$$

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

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

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

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 then Np-237. It also requires the decay of U-237 with its half life of 6.75 days. 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 shutdown 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.

The range of applicability will be described by four parameters: burnup, spectrum, enrichment, and specific power. These four parameters will cover the major independent ways of creating errors in isotopic concentrations.

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-236, U-238, Pu-239, Pu-240, or Pu-242).

Only the four parameters; burnup, spectrum, enrichment, and specific power will be used to establish the range of applicability. Table 2-3 represents the range of applicability that can be supported by the data. Extrapolation of data range via trending analysis, presented in Section 2.3, is permitted in ANSI/ANS-8.1 and is used here. 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.

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

where,

$\bar{u}_i$ = 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 - 50
Spectral Index*	15.6 - 16.5	15 - 17
Initial Enrichment (Wt. % U-235)	2.45 - 3.90	0.71 - 5.0
Specific Power(MW/MTU)	10.7 - 39.6	0 - 60

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

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:

Eq. 2-5

$$x = \frac{M}{C}$$

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 follow:

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

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 , 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 $_4$b and $_i$B$_i$*P 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]

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

where,

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

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:

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

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

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

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:

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

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:

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

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.[2-13] 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 nine isotopes with respect to burnup, burnup*spectrum, and burnup*enrichment. 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-236, 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.

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

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

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

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

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

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

Table 2-4 provides the mean biases, uncertainties, and the correction factors (f_{buc}) for each selected actinide. 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) validation of SAS2H/27BURNUPLIB of the SCALE 4.2 computer code package. It is the purpose of this chapter to seek the acceptance 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. Using this specific code, data, and methodology, a set of isotopic correction factors was developed that can be used directly in modifying future calculated inventory of the selected actinides.

Therefore, this section seeks NRC acceptance of the following:

- That 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.

- That 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.

- That the SAS2H sequence of the SCALE 4.2 code system using 27BURNPLIB cross sections has been validated and that appropriate isotopic correction factors have been determined.

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	0.186	0.814	0.814
*U-235	54	1.025	0.052	1.084	1.084
U-236	53	1.000	0.064	0.936	0.936
U-238	48	1.000	0.009	0.991	0.991
Pu-238	40	1.000	0.134	0.866	0.866
*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-236, U-238, and Pu-238 are 0.973, 0.999, 1.001, 1.005, respectively.

3. CRITICALITY VALIDATION

The design of spent fuel packages using burnup credit imposes unique requirements for criticality analysis method validation. Many components of the burnup credit criticality analysis method proposed in this topical report have been widely used in the design and licensing of conventional transportation and storage system packaging using the fresh fuel assumption. These methods must be shown to be valid for use in burnup credit applications. American National Standards ANSI/ANS-8.1[3-1] and 8.17[3-2] provide criteria for nuclear criticality safety in away-from-reactor spent fuel handling, storage, and transportation activities. ANSI/ANS-8.1 applicability is specifically focused on criteria for validating analysis methodologies. Validation consistent with ANSI/ANS-8.1 requires:

- Use of "adequate" calculational techniques and nuclear data
- Correlation of analytical results with experimental data to establish analysis method bias
- Establishment of reactivity margin consistent with uncertainties.

ANSI/ANS-8.1 also requires documentation of validation results and allows for the use of trends to extend the criticality analysis method bias beyond the range of experiment conditions evaluated to establish the bias. The burnup credit criticality analysis validation described in this report follows the guidance provided in ANSI/ANS 8.1 and 8.17.

The criticality analysis method validation process described in this chapter is intended to be generically applicable to any criticality analysis code system. Criticality analysis validation is accomplished primarily through correlation of analytical results to benchmark critical experiments for systems containing UO_2 and mixed oxide fuel (MOX). A representative set of benchmark experiments suitable for use in burnup credit method validation is presented, and a procedure for combining benchmark calculation results and deriving a subcritical safety limit is described.

The validation process is demonstrated with specific codes and nuclear data libraries. Criticality analysis calculational techniques and nuclear data used in this chapter are included in Version 4.2 of the SCALE modular code system.[3-3] SCALE is a well-established code system that has been used widely in away-from-reactor applications for criticality safety analyses via its CSAS (Criticality Safety Analysis Sequence) analysis sequences. CSAS sequences invoke standardized procedures to provide appropriate neutron cross sections for use in criticality (k_{eff}) calculations. Cross section processing is performed by the SCALE 4.2 NITAWL-II and BONAMI modules. Where specific problems require cell-weighting of cross sections prior to the criticality calculation, the CSAS2X sequence is used in which the 1-D discrete ordinates code XSDRNPM is invoked. Criticality calculations are performed via three-dimensional (3-D) Monte Carlo calculations using KENO V.a. The CSAS sequences, and the 27BURNUPLIB neutron cross section library (henceforth referred to as the CSAS/27BURNUPLIB code systems), are used to demonstrate the generic validation process.

In summary, this chapter presents, for review and approval, a generic validation methodology for actinide-only burnup credit criticality calculational methods. The validation methodology includes: 1) the selection of a set of benchmark critical experiments for spent nuclear fuel package design, 2) the trending analyses of the calculated multiplication factors against selected parameters, and 3) the determination of an acceptance criterion for the calculated maximum value of k_{eff} referred to as an upper safety limit. Based on the validation results presented in this chapter, the review and approval of the validation of the CSAS/27BURNUPLIB code system for actinide-only burnup credit are requested.

3.1 CRITICALITY VALIDATION DATA

Current practice for validating criticality safety computational methods relies primarily on measured data from well characterized critical experiments using fresh fuel. Available benchmark data include a diverse set of low-enrichment heterogeneous rod lattice systems designed to test computational method capabilities over a wide range of parameters important to criticality safety. A large number of UO_2 critical experiments have been performed that are useful in validating specific non-burnup related parameters, such as variations in rod pitch, moderation, presence of soluble neutron absorbers, reflector effects, and external fixed and integral neutron absorbers. Criticality benchmark experiments containing MOX fuel are also available for validating the effect of higher order actinides on spent nuclear fuel system reactivity.

3.1.1 Validation Experiments

Burnup credit criticality analysis methods proposed for use in spent nuclear fuel package design should be validated against a diverse set of critical experiments covering the range of design features and operating conditions anticipated for the specific package design application. A reference set of 57 critical experiments has been selected to demonstrate the generic validation methodology. This generic reference set of experiments measures method accuracy over a wide range of conditions for systems containing low enrichment, fresh fuel rod lattices similar to light water reactor fuel assemblies. The experiment set includes 19 UO_2, 2 UO_2-Gadolinium, and 36 MOX configurations. A brief description of the experiments and effective neutron multiplication factor (k_{eff}) results calculated using the CSAS/27BURNUPLIB code system are presented in Table 3-1. More detailed information can be found in the original references listed or the technical reports generated in support of this project.[3-15, 3-16]

3.1.1.1 UO_2 Fuel Critical Experiments

The 19 UO_2 critical experiments from eight references are selected to examine six different aspects of spent nuclear fuel package criticality:

- Neutron interaction between PWR type fuel assemblies
- Effectiveness of neutron flux traps between fuel assemblies
- Effect of voiding on the effectiveness of flux traps

- Effectiveness of neutron absorber plates and rods to reduce interaction between fuel assemblies
- Reactivity effect of commonly used package shielding materials
- Neutron spectra shift or relative neutron moderation caused by dissolved boron.

Table 3-1 lists the experiments and their distinctive characteristics as related to the aspects of criticality listed above. The experiments are water moderated and reflected, unless otherwise noted.

The experiments included in the UO_2 experiment subset are typical of experiments used in fresh fuel assumption package applications. The specific experiments included in this subset were selected to cover a broad range of features found in SNF package designs. Five experiments are specifically included as comparison cases for UO_2-Gadolinium and MOX experimental arrangements to identify potential trends. The UO_2 subset is designed to provide a comprehensive test of the neutronics calculational ability. The specific UO_2 experiments included in Table 3-1 are proposed as an integral component of the generic reference set of 57 critical experiments. However, specific package design applications may not incorporate features included in the UO_2 subset (e.g., depleted uranium shielding) or may incorporate features not included in the subset (e.g., hafnium supplemental absorber materials). Package designers should confirm that the generic reference set covers the significant criticality control package design features prior to applying the UO_2 subset without modification.

3.1.1.2 UO_2-Gadolinium Critical Experiments

Spent fuel contains significant neutron absorbing isotopes that result in "hardening" of the neutron spectrum relative to nuclear fuel systems containing fresh fuel only. The spectrum hardening, or the increase in average neutron energy, results from increased competition for thermal neutrons within the system due to the presence of fission product and higher order actinide neutron absorbers which are not present in fresh fuel systems. The addition of large thermal neutron absorption cross section isotopes, such as Gd-155, in significant concentrations integral with the UO_2 fuel matrix, can also result in hardening of the neutron spectrum relative to pure UO_2 systems of comparable initial enrichment.

The UO_2-gadolinium critical experiments are included in the set of critical experiments to account for the effects that may arise because of neutron spectrum hardening in UO_2 systems. Table 3-1 presents calculated average lethargy for absorption values for each critical experiment; for the two UO_2-Gd_2O_3 experiments, relatively low values are observed, denoting the spectrum hardening due to the inclusion of gadolinium. Nevertheless, since these experiments represent fresh fuel with a small variation (inclusion of Gd_2O_3), they are considered part of the UO_2-only subset. These experiments are applicable to any analysis that references this topical report as a basis for actinide-only burnup credit criticality analysis method validation.

Table 3-1. Critical Benchmark Experiments for Burnup Credit Method Validation with CSAS/27BURNUPLIB Calculated Results

Experiment Case	Enrichment U-235 (wt %)	Description	Original Reference[1]	Lattice Water/Fuel Volume	ALA[2]	k_{eff} ± std. dev.
LWR UO_2 Fuel Pin Lattices - 19 Experiments						
Absorber plates:						
Experiment 1	2.35	Exp. 005: No Absorber (H_2O)	Ref. 3-4	2.92	19.0	0.9908±0.0013
Experiment 2	2.35	Exp. 017: Boral	Ref. 3-4	2.92	18.9	0.9930±0.0010
Experiment 3	2.35	Exp. 024: Aluminum plates	Ref. 3-4	2.92	19.0	0.9925±0.0014
Experiment 4	2.35	Exp. 028: Stainless steel	Ref. 3-4	2.92	19.0	0.9936±0.0014
Reflecting walls:						
Experiment 5	4.31	Uranium walls	Ref. 3-5	1.60	17.4	0.9980±0.0016
Experiment 6	4.31	Lead walls	Ref. 3-5	1.60	18.2	0.9978±0.0016
Experiment 7	4.31	Steel walls	Ref. 3-6	1.60	17.9	0.9979±0.0016
Soluble Boron:						
Experiment 8	4.31	Exp. 173: no boron	Ref. 3-7	1.59	18.3	0.9947±0.0011
Experiment 9	4.31	Exp. 177: 2550 ppmb	Ref. 3-7	1.59	17.3	0.9987±0.0012
Experiment 10	4.31	Exp. 178: no boron	Ref. 3-7	1.09	17.7	0.9950±0.0011
Experiment 11	4.31	Exp. 181: 2550 ppmb	Ref. 3-7	1.09	16.7	0.9881±0.0013
Flux traps:						
Experiment 12	4.31	Exp. 214R: flux traps (no voids)	Ref. 3-8	1.60	17.5	0.9943±0.0016
Experiment 13	4.31	Exp. 214V3: flux traps (with voids)	Ref. 3-8	1.60	17.4	0.9933±0.0010
UO_2 triangular lattices:						
Experiment 14	2.35	EPRI MOX Comparison	Ref. 3-11	1.20	17.9	0.9909±0.0013
Experiment 15	2.35	EPRI MOX Comparison	Ref. 3-11	3.69	19.2	0.9960±0.0013
UO_2 square lattices:						
Experiment 16	2.46/4.02	UO_2/Gd_2O_3 Comparison	Ref. 3-10	1.87	17.5	0.9930±0.0011
Experiment 17	5.74	SAXTON MOX Comparison	Ref. 3-12	1.93	18.2	0.9931±0.0019
Experiment 18	5.74	SAXTON MOX Comparison	Ref. 3-12	5.07	19.2	0.9955±0.0012
3x3 assy arrays w/absorbers:						
Experiment 19	2.46	Core IV: 84 B_4C pins-1 pitch between assemblies	Ref. 3-9	1.84	17.8	0.9898±0.0010
UO_2-Gadolinium Lattices - 2 Experiments						
UO_2/Gd_2O_3 fuel rods:						
Experiment 20	1.94/2.46/4.02	Core 14: 12 Gd fuel rods	Ref. 3-10	1.88	17.5	0.9905±0.0011
Experiment 21	1.94/2.46/4.02	Core 16: 16 Gd fuel rods	Ref. 3-10	1.88	17.5	0.9923±0.0011

Experiment Case	Enrichment U-235 (wt %)	Description	Original Reference[1]	Lattice Water/Fuel Volume	ALA[2]	$k_{eff} \pm$ std. dev.
LWR Mixed Oxide Criticals-36 Experiments						
EPRI UO_2/PuO_2 2wt%PuO_2 7.8wt%Pu-240:						
Experiment 22	0.71	0.700-in. pitch, 0 ppmb	Ref. 3-11	1.20	17.5	0.9960±0.0014
Experiment 23	0.71	0.700-in. pitch, 688 ppmb	Ref. 3-11	1.20	17.1	0.9977±0.0013
Experiment 24	0.71	0.870-in. pitch, 0 ppmb	Ref. 3-11	2.53	18.6	1.0032±0.0011
Experiment 25	0.71	0.870-in. pitch, 1090 ppmb	Ref. 3-11	2.53	18.1	1.0050±0.0013
Experiment 26	0.71	0.990-in. pitch, 0 ppmb	Ref. 3-11	3.64	18.9	1.0048±0.0014
Experiment 27	0.71	0.990-in. pitch, 767 ppmb	Ref. 3-11	3.64	18.6	1.0073±0.0009
SAXTON UO_2/PuO_2 6.6wt%PuO_2 8.6wt%Pu-240:						
Experiment 28	0.71	0.52-in. pitch	Ref. 3-12	1.68	17.5	1.0025±0.0012
Experiment 29	0.71	0.56-in. pitch	Ref. 3-12	2.16	17.9	1.0035±0.0017
Experiment 30	0.71	0.56-in. pitch, 337 ppmb	Ref. 3-12	2.16	17.7	0.9998±0.0016
Experiment 31	0.71	0.735-in. pitch	Ref. 3-12	4.70	18.8	1.0046±0.0017
Experiment 32	0.71	0.792-in. pitch	Ref. 3-12	5.67	19.0	1.0063±0.0017
Experiment 33	0.71	1.04-in. pitch	Ref. 3-12	10.75	19.3	1.0076±0.0016
PNL4976 MOX and UO_2 2wt%PuO_2 7.9wt%Pu-240:						
Experiment 34	MOX 0.71 UO_2 4.31	MOX and UO_2 rods in uniform pattern	Ref.3-13	0.49	15.5	0.9864±0.0012
PUP UO_2/PuO_2 triangular 2wt%PuO_2 8wt%Pu-240:						
Experiment 35	0.71	0.80-in lattice spacing	Ref. 3-14	1.211	17.9	0.9882±0.0010
Experiment 36	0.71	0.93-in lattice spacing	Ref. 3-14	1.987	18.6	0.9960±0.0011
Experiment 37	0.71	1.05-in lattice spacing	Ref. 3-14	2.808	18.9	0.9909±0.0010
Experiment 38	0.71	1.143-in lattice spacing	Ref. 3-14	3.513	19.1	1.0003±0.0010
Experiment 39	0.71	1.32-in lattice spacing	Ref. 3-14	5.019	19.3	1.0034±0.0010
Experiment 40	0.71	1.386-in lattice spacing	Ref. 3-14	5.635	19.3	1.0005±0.0010
PUP UO_2/PuO_2 triangular 2wt%PuO_2 16wt%Pu-240:						
Experiment 41	0.71	0.93-in lattice spacing	Ref. 3-14	1.987	18.5	0.9997±0.0010
Experiment 42	0.71	1.05-in lattice spacing	Ref. 3-14	2.808	18.8	0.9983±0.0010
Experiment 43	0.71	1.143-in lattice spacing	Ref. 3-14	3.513	19.0	1.0032±0.0010
Experiment 44	0.71	1.32-in lattice spacing	Ref. 3-14	5.019	19.2	1.0025±0.0009
PUP UO_2/PuO_2 triangular 2wt%PuO_2 24wt%Pu-240:						
Experiment 45	0.71	0.80-in lattice spacing	Ref. 3-14	1.211	17.7	0.9902±0.0009
Experiment 46	0.71	0.93-in lattice spacing	Ref. 3-14	1.987	18.4	0.9949±0.0009
Experiment 47	0.71	1.05-in lattice spacing	Ref. 3-14	2.808	18.8	0.9985±0.0009
Experiment 48	0.71	1.143-in lattice spacing	Ref. 3-14	3.513	19.0	1.0023±0.0009
Experiment 49	0.71	1.32-in lattice spacing	Ref. 3-14	5.019	19.2	1.0051±0.0009
Experiment 50	0.71	1.386-in lattice spacing	Ref. 3-14	5.635	19.2	1.0072±0.0009

Table 3-1. Critical Benchmark Experiments for Burnup Credit Method Validation with CSAS/27BURNUPLIB Calculated Results (Continued)

Experiment Case	Enrichment U-235 (wt %)	Description	Original Reference[1]	Lattice Water/Fuel Volume	ALA[2]	k_{eff} ± std. dev.
PUP UO_2/PuO_2 triangular 4wt%PuO_2 18wt%Pu-240:						
Experiment 51	0.71	0.85-in lattice spacing	Ref. 3-14	1.500	18.0	0.9982±0.0010
Experiment 52	0.71	0.93-in lattice spacing	Ref. 3-14	1.993	18.4	0.9945±0.0011
Experiment 53	0.71	1.05-in lattice spacing	Ref. 3-14	2.815	18.8	1.0007±0.0011
Experiment 54	0.71	1.143-in lattice spacing	Ref. 3-14	3.521	19.0	1.0007±0.0011
Experiment 55	0.71	1.386-in lattice spacing	Ref. 3-14	5.647	19.2	1.0082±0.0010
Experiment 56	0.71	1.60-in lattice spacing	Ref. 3-14	7.859	19.4	1.0086±0.0009
Experiment 57	0.71	1.70-in lattice spacing	Ref. 3-14	9.000	19.4	1.0096±0.0009
Total Number of Experiments = 57						

[1] This table summarizes experiments analyzed and documents results provided in Reference 3-16.
[2] Average Lethargy for Absorption (ALA). Used to assess relative energy spectra for each critical configuration.

3.1.1.3 Mixed-Oxide Critical Experiments

Spent fuel contains many actinides and fission products that are not present in fresh fuel. In addition to the U-235 and U-238 isotopes present in spent fuel, the set of burnup credit isotopes considered in this topical report includes other actinide isotopes that are important in the neutronics modeling of spent fuel systems. Important actinide isotopes include the fissile isotopes Pu-239 and Pu-241, and major neutron absorbing isotopes such as Pu-240. Although laboratory critical experiment measured k_{eff} data are not available for irradiated nuclear fuel, experiments have been performed with mixed oxide (MOX) fuel. Thirty-six MOX fuel critical experiments have been selected to use for validating burnup credit analysis methodologies and are included in the benchmark set. Four sources of MOX critical experiments are used. In addition, four UO_2 critical experiments are included from two of the MOX critical experiment series to identify any trends that may exist between UO_2 and MOX fuel in similar configurations.[3-11, 3-12]

The 36 MOX experiments are included in the reference set of benchmark experiments to provide criticality analysis method validation data for the selected actinide-only burnup credit isotopes. Since the UO_2 and UO_2-Gadolinium experiments provide benchmark data for U-235 and U-238 only, the MOX experiments are necessary to provide validation data for the other fissile and neutron absorbing actinide isotopes. The MOX experiments are applicable to any analysis that references this topical report as a basis for actinide-only burnup credit criticality analysis method validation.

3.1.2 Range of Validation Experiment Conditions

The range of material compositions and geometric arrangements representative of conditions expected in a spent fuel package must be characterized to establish requirements for the burnup credit criticality analysis method validation. Column 1 of Table 3-2 summarizes key criticality analysis parameters. The range of conditions that can be anticipated for spent fuel packages is provided in column 2 of the table. Two sets of critical experiments (i.e., UO_2 and MOX) from several experimental sources have been selected for burnup credit criticality analysis method validation. Column 3 of Table 3-2 provides the range of spent fuel composition and physical system characteristics that are covered by the critical experiments selected. The experiments cover a wide range of fuel compositions and anticipated spent fuel package physical conditions.

Validation of burnup credit criticality analysis methodologies requires consideration of a larger number of isotopes than is necessary with the fresh fuel assumption. In addition to the uranium in spent fuel, the set of burnup credit isotopes considered in this topical report include significant actinide isotopes. The set of 36 MOX experiments selected provides the experimental measurement data necessary to validate burnup credit analysis method treatment of all the actinide isotopes included in the methodology. The MOX fuel benchmark experiments are strongly influenced by the key fissile isotopes (U-235, Pu-239, and Pu-241) and major neutron absorbing isotopes (U-238 and Pu-240). Figures 3-1 and 3-2 provide a comparison of fuel compositions for each MOX validation experiment series relative to spent fuel compositions for two levels of fuel burnup. The five key actinides listed above, in addition to U-234, Pu-238, Pu-242 and Am-241, are represented adequately in the MOX benchmarks. The concentrations of U-236 in the experiments are not comparable to the representative SNF values; nevertheless, it has been demonstrated that it is acceptable to include this isotope in criticality calculations.[3-16]

3.1.3 Qualification of Data

The critical experiments included in the generic reference set are recognized "benchmark standard" experiments performed specifically for reactor core design and criticality analysis method benchmarking purposes. Each experiment has been formally documented in reports issued by the organizations involved in the measurements. These reports were reviewed prior to publication, and have been further reviewed by their intended audience. Finally, the number of criticality calculations performed at the time of the experiments and subsequently by various organizations and individuals applying the data in validation work serves as a confirmation of the measurements under the specified conditions. Many of the UO_2 benchmark experiments included in the reference set have been previously used and accepted for such purposes in numerous fresh fuel storage and transportation package design and licensing applications. In addition, the use of a comprehensive set of experimental data from a number of independent experimental facilities provides a high degree of assurance that potential inaccuracies, errors, or omissions in any one experiment or group of experiments will not significantly affect the overall criticality analysis method bias result derived from the application of the entire set of experimental data.

Table 3-2. Areas of Applicability Matrix

Key Parameters	Anticipated SNF Package Conditions	57 Critical Experiments
Fuel Rod Parameters		
Fuel Composition Isotopic Composition	Spent Fuel	U-234 U-235 U-236 U-238 Pu-238 Pu-239 Pu-240 Pu-241 Pu-242 Am-241
Burnup	0 to 56 GWd/MTU	Unirradiated UO_2 and MOX fuel
Initial Enrichment (wt. % U-235)	0.71 to 5.00%	UO_2: 2.35 to 5.74% UO_2 - Gd_2O_3: 1.94 to 4.02% MOX: 0.71 to 4.31% (wt % U-235) 2 to 6.6% (wt% PuO_2)
Cooling Time	5 to 100 Years	N/A
Fuel Material Nuclear Properties Fuel Temperature	70°F	70°F
Fuel Material Form	Fuel: Irradiated UO_2 Cladding: Zircaloy Stainless Steel	Fuel: UO_2 UO_2 - Gd_2O_3 MOX Cladding: Zircaloy Stainless Steel Aluminum
Fuel Material Density	10.0 to 10.4 g/cm^3 (91% to 95% of theoretical density)	UO_2: 9.2 to 10.4 g/cm^3 UO_2 - Gd_2O_3: 9.5 to 10.2 g/cm^3 MOX: 9.5 to 10.4 g/cm^3
Fuel Rod Geometry	Square lattice, heterogeneous 0.71 to 0.99 cm pellet dia 0.005 to 0.017 cm gap 0.79 to 1.12 cm cladding OD	Square and triangular lattices, heterogeneous 0.86 to 1.28 cm pellet dia 0 to 0.009 cm gap 0.99 to 1.44 cm cladding OD
Fuel Rod Spacing	1.07 to 1.47 cm pitch	UO_2: 1.42 to 2.21 cm pitch UO_2 - Gd_2O_3: 1.64 cm pitch MOX: 1.32 to 4.32 cm pitch

Key Parameters	Anticipated SNF Package Conditions	57 Critical Experiments
Array Parameters		
Fixed Neutron Absorbers	Borated materials (e.g., B_4C, borated stainless steel, boral, etc.)	External B_4C rods Boral, stainless steel, and aluminum plates
Materials of Construction within Array	Guide tubes	Water gaps
Moderator Conditions		
Water Density	1 g/cm^3	1 g/cm^3
Water Temperature	70°F	70°F
Moderator to Fuel Volume Ratio	1.3 to 1.9	UO_2: 1.09 to 5.07 UO_2 - Gd_2O_3: 1.88 MOX: 0.49 to 10.75
Soluble Boron Concentration	0 ppm	UO_2: 0 to 2550 ppm UO_2 - Gd_2O_3: 1579 to 1654 MOX: 0 to 1090 ppm
Reflector and Interaction Conditions		
Reflector Composition	Water, depleted uranium and stainless steel reflectors	Water, depleted uranium, stainless steel and lead reflectors
Interaction with other Fissile Material	Fissile uranium and plutonium isotopes in SNF	Fissile uranium and plutonium isotopes in MOX fuel pins

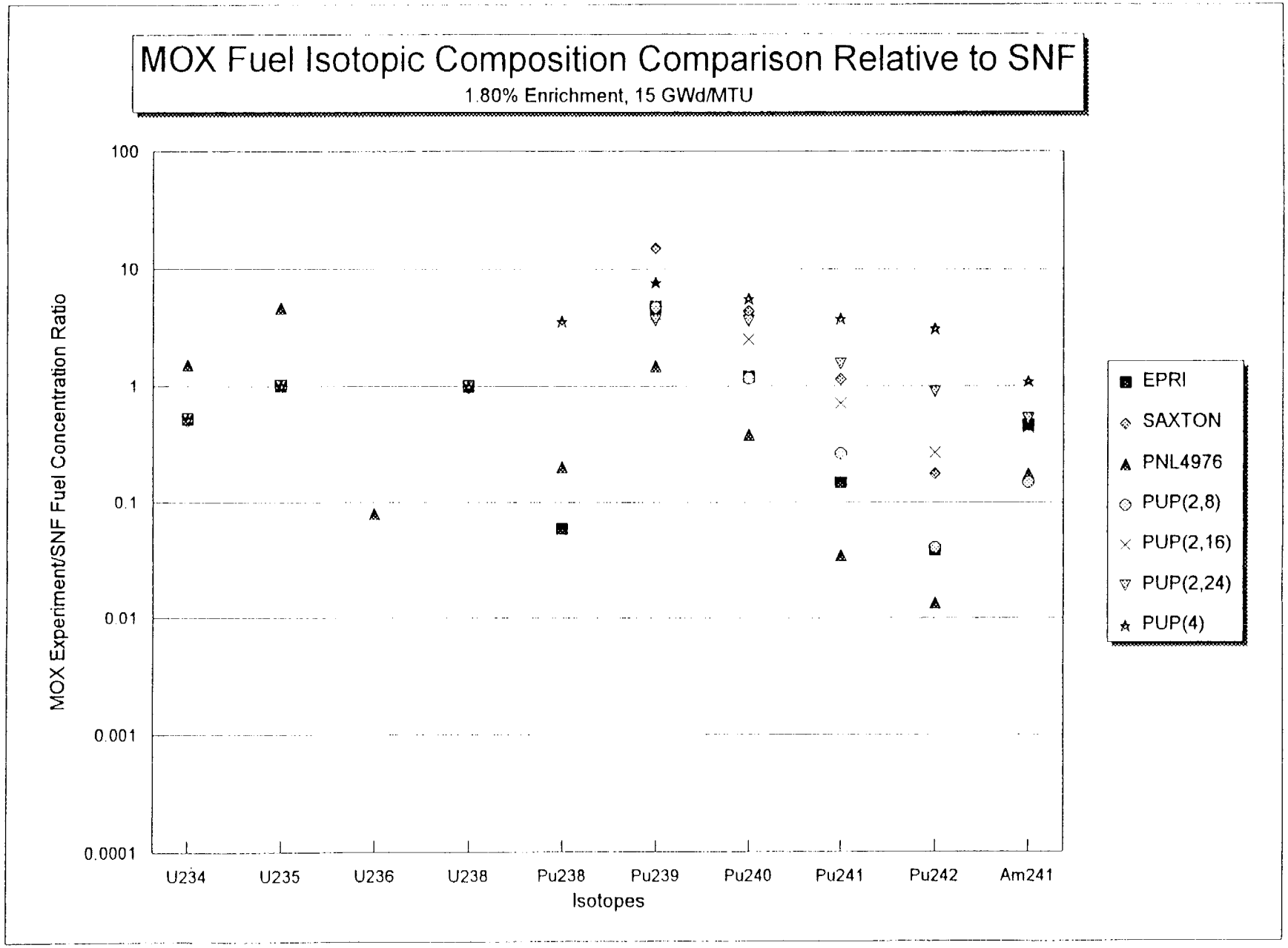


Figure 3-1. MOX Benchmark Experiment Composition Summary (against Low Burnup)

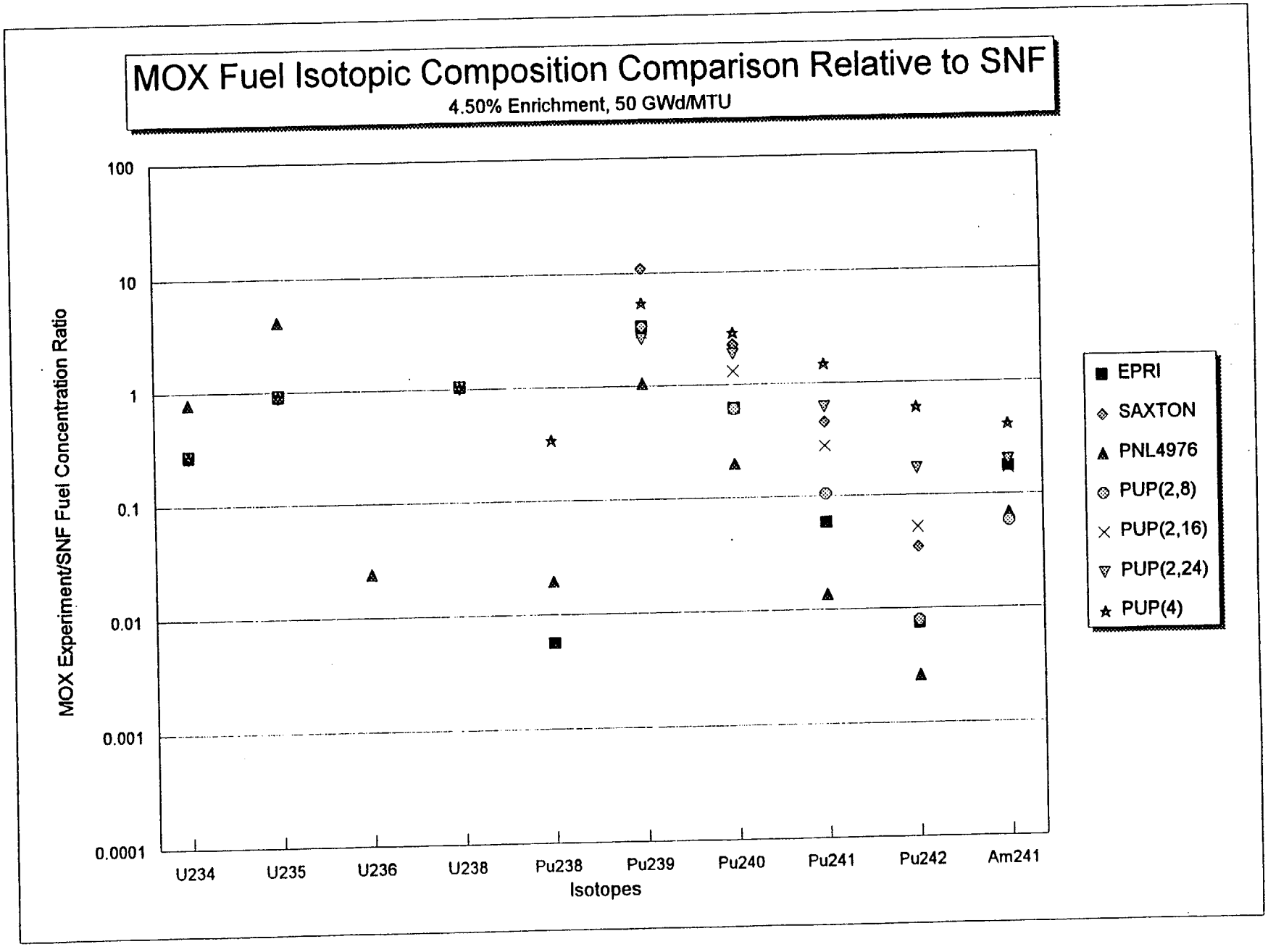


Figure 3-2. MOX Benchmark Experiment Composition Summary (against High Burnup)

3.2 CALCULATIONAL BIAS AND UNCERTAINTIES

Specific guidance for use of calculational methods in the analysis of LWR fuel is provided in ANSI/ANS-8.1. The Standard specifies that "bias shall be established by correlating the results of criticality experiments with results obtained for these same systems by the method being validated." Based on this Standard, this section describes the recommended approach and statistical methods for determining the bias in the calculation of k_{eff}, along with uncertainties associated with that bias.

3.2.1 ANSI/ANS-8.17 Guidance

The approach to be taken in demonstrating subcriticality based on the numerical calculation of the effective neutron multiplication factor is prescribed in Section 5.1 of ANSI/ANS 8.17. The following paragraphs describe the recommended approach as set forth in the Standard. The criterion to establish subcriticality requires that the calculated multiplication factor, k_s, be less than or equal to an established maximum allowable multiplication factor based on benchmark calculations and uncertainty terms, i.e.,

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad \text{Eq. 3-1}$$

where

k_s = calculated allowable maximum multiplication factor, k_{eff}, for the system being evaluated

k_c = the mean value of k_{eff} resulting from the calculation of benchmark criticality experiments using a specific calculational method

Δk_s = uncertainty in the value of k_s

Δk_c = uncertainty in the value of k_c

Δk_m = an arbitrary administrative margin to ensure subcriticality.

ANSI/ANS-8.17 provides additional detailed guidance describing analysis considerations included in each term of Equation 3-1. The uncertainty in method bias, Δk_c, connected with the calculation of k_c, may include uncertainties in the critical experiments, statistical and/or convergence uncertainties in the benchmark calculations, uncertainties due to extrapolation beyond the range of experimental data, and uncertainties due to limitations or weaknesses in the geometrical or nuclear modeling of the critical experiments. Similarly, for a given subcritical system, there is the uncertainty Δk_s associated with the calculated k_{eff} value for the system, k_s. This uncertainty includes any statistical/convergence uncertainty related to the analysis method (i.e., Monte Carlo uncertainties) and modeling uncertainties related to basket construction (i.e., material composition, material thickness, and fabrication tolerances) not accounted for with worst-case treatments in the computational model. Each of the various uncertainties is combined statistically if independent, or combined additively if statistically correlated.

3.2.2 Bias and Uncertainty From Critical Experiments

The criticality analysis method bias is calculated as a direct additive measure of systematic disagreement between the calculated and the measured experimental data for direct application of the ANSI/ANS-8.17 subcriticality safety criteria. The ANSI/ANS-8.17 form of the subcriticality safety criteria, Equation 3-1, is rearranged to define the criticality analysis method bias and bias uncertainty in more directly recognizable terms. If the method bias is defined as β, where $\beta = 1 - k_c$, and the uncertainty in bias is defined as $\Delta\beta$, where $\Delta\beta = \Delta k_c$, the subcriticality safety criteria can be rewritten as follows:

$$k_s \leq 1.0 - \beta - \Delta k_s - \Delta\beta - \Delta k_m \qquad \text{Eq. 3-2}$$

The value k_c and thus the method bias β are not necessarily a constant over the full range of variable parameters of interest. If trends exist that cause the benchmark values of k_{eff} to vary with one or more parameters (e.g., enrichment, fuel-to-moderator ratio, etc.), then β is most appropriately determined from a best fit for the calculated k_{eff} values, as a function of the parameter upon which it is dependent. A statistical approach is presented in the following subsection that can be applied to perform trend analyses on benchmark calculation results and calculate a method bias as a function of a single parameter. A method for determining the uncertainty, $\Delta\beta$, connected with the calculation of β is also presented.

3.2.2.1 Lower Prediction Band Technique

Based on the criteria for subcriticality set forth in ANSI/ANS 8.17, a statistical technique has been developed for the determination of subcritical limits using the prediction interval method,[3-17] which was also used for the correction factors in Chapter 2. Similar statistical techniques have been previously applied for the determination of subcritical limits in validation studies for the CSAS/27BURNUPLIB code system.[3-18, 3-19] This approach is a single-sided statistical method for the determination of an upper safety limit (USL) based on the statistical analysis of a number of critical systems. The USL is determined such that there is a high degree of confidence that a calculated result is subcritical; a system is considered acceptably subcritical if a calculated k_{eff} plus calculational uncertainties lies at or below this limit (i.e., $k_s + \Delta k_s \leq$ USL). Thus, based on Equation 3-2, the USL is the statistically determined magnitude of the sum of the biases, uncertainties, and administrative safety margin computed for a set of critical benchmarks, such that with a high degree of confidence,

$$\text{USL} \leq 1.0 - \beta - \Delta\beta - \Delta k_m \qquad \text{Eq. 3-3}$$

Based on a given set of critical experiments, the USL can be determined as a function of key system parameters, such as the average energy group causing fission (AEG), average lethargy for absorption (ALA), fuel enrichment, or fuel/moderator ratio. Because both β and $\Delta\beta$ can vary with a given parameter, the USL is typically expressed as a function of the parameter. This approach is conceptually illustrated in Figure 3-3. In this figure, the upper line [$k_c(x)$] represents a linear regression fit of a set of benchmark experiment calculation results plotted as a function

of the trended parameter, x. The difference between the linear regression fit and k_{eff} = 1.0 is the calculational bias. The middle line [$k_c(x)$- $\Delta k_c(x)$] represents the lower prediction band (LPB) for a single additional calculation; e.g., the user can be 95% confident ($1-\gamma_1=0.95$) that the next calculated value of k_{eff} for a critical experiment will be greater than [$k_c(x)-\Delta k_c(x)$]. The prediction band is determined statistically based on the existing data and a specified level of confidence; the greater the standard deviation in the data or the larger the confidence desired, the larger the band width will be. The prediction band accounts for uncertainties in the experiments, calculational approach, and calculational data (e.g., neutron cross sections), and is therefore a statistical basis for $\Delta\beta$, the uncertainty in the value of the bias, β. The bottom line in the figure represents the upper safety limit for subcriticality, based on an additional margin of subcriticality. This safety margin provides further assurance of subcriticality and represents the quantity Δk_m defined earlier.

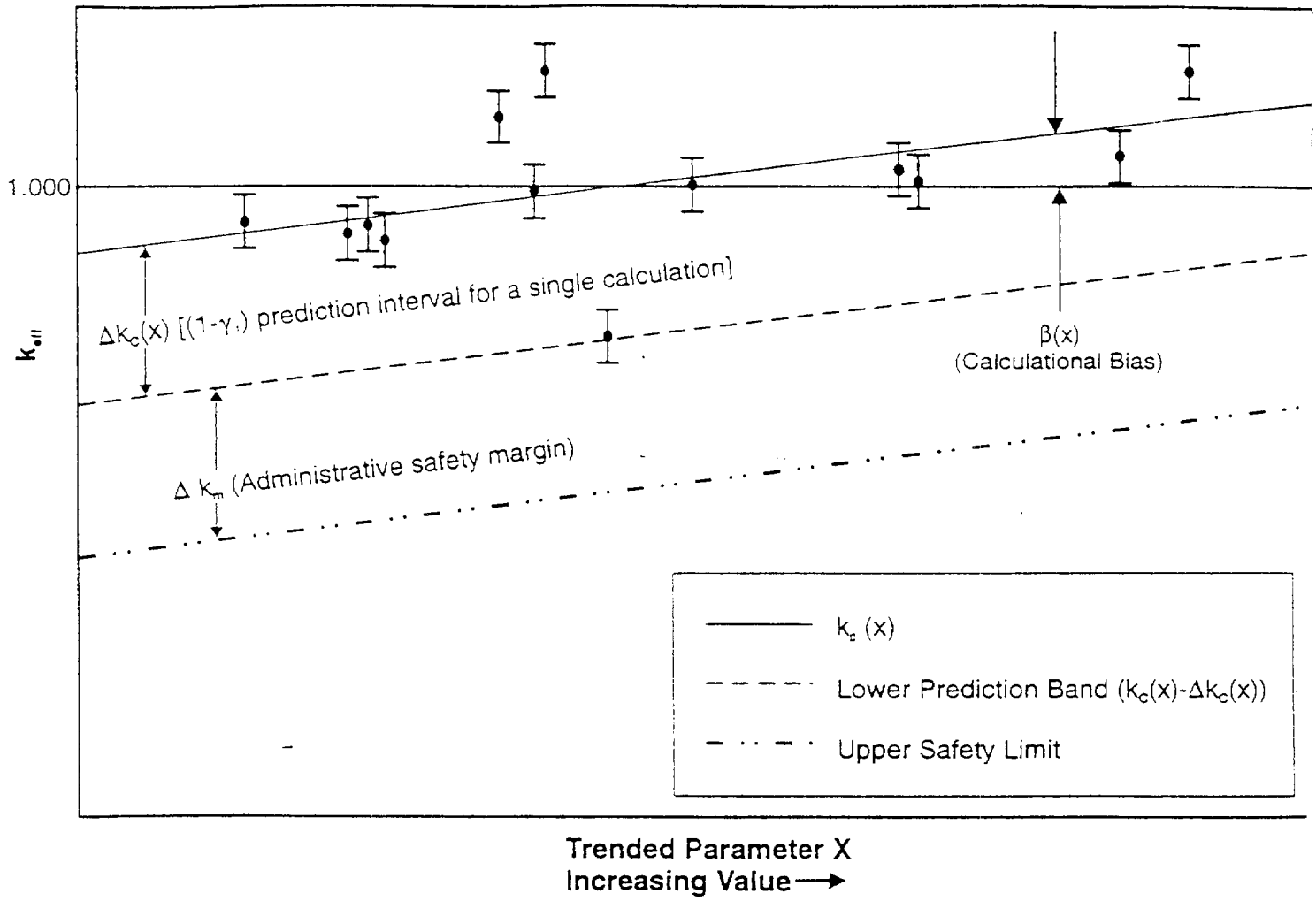


Figure 3-3. Illustration of the Lower Prediction Band Technique for the Determination of an Upper Safety Limit (USL)

Based on Equation 3-3 and the definitions provided above, the limiting USL condition (i.e., USL $= 1.0 - \beta - \Delta\beta - \Delta k_m$) is defined as:

$$USL = 1.0 - \overline{\beta} - \Delta\beta - \Delta k_m \qquad \text{Eq. 3-4}$$

for cases when the calculated multiplication factors do not show a trend with any parameter ($\overline{\beta}$ is the mean bias), or

$$USL(x) = 1.0 - \beta(x) - \Delta\beta(x) - \Delta k_m \qquad \text{Eq. 3-5}$$

when the calculated values show a statistically significant trend against parameter x. In the unlikely event that the calculated values show trends against more than one parameter, multiple linear regression would have to be employed to compute both the bias and uncertainty as a function of the multiple parameters. To determine if the calculated values exhibit a significant trend against parameter x, the data is fitted through linear regression to obtain the equation

$$k_c(x) = a + b\,x \qquad \text{Eq. 3-6}$$

and a slope test[3-20] (variation of the Student's t test) is performed. The test requires obtaining the test statistic, T,

$$T = b\sqrt{\frac{(n-2)\,S_{xx}}{SS_R}} \qquad \text{Eq. 3-7}$$

where

$$S_{xx} = \sum_{i=1,n} (x_i - \bar{x})^2 \qquad \text{Eq. 3-8}$$

and

$$SS_R = \sum_{i=1,n} (k_i - a - bx_i)^2 \qquad \text{Eq. 3-9}$$

The test statistic is then compared to the Student's t-distribution with 95% ($\gamma = 0.05$) confidence and n-2 degrees of freedom. Given a null hypothesis of "no statistically significant trend exists (slope is zero)," the hypothesis is accepted for $|T| < t_{\gamma/2,n-2}$, and rejected otherwise.

The bias, β, is treated as a function of zero, one, or multiple parameters, based on the linear regression and trend testing on the calculated multiplication factors. Figure 3-3 presents the case when the bias depends on one trending parameter, $\beta(x) = 1.0 - k_c(x)$.

Next, the prediction band for an additional calculation, $\Delta\beta$, may be determined using the relationship:[3-21]

$$\Delta\beta = t_{\gamma_1,n-1}\, s_p \left[1 + \frac{1}{n}\right]^{1/2} \qquad \text{Eq. 3-10}$$

for cases with no trend, and

$$\Delta\beta(x) = t_{\gamma_1,n-2}\, s_p \left[1 + \frac{1}{n} + \frac{(x-\bar{x})^2}{S_{xx}}\right]^{1/2} \qquad \text{Eq. 3-11}$$

for the one trend cases, where

n = the number of critical calculations used in establishing $k_c(x)$

$t_{\gamma 1,m}$ = the Student-t value for γ_1 and m degrees of freedom

$\bar{x}$ = the mean value of parameter x in the set of calculations

s_p = the pooled standard deviation for the set of criticality calculations.

The pooled standard deviation is obtained from the pooled variance, s_p^2. Pooled variance is given by:

$$s_p^2 = s^2 + s_w^2 \qquad \text{Eq. 3-12}$$

for the no trend cases, and

$$s_p^2 = s_{k(x)}^2 + s_w^2 \qquad \text{Eq. 3-13}$$

for the one trend cases. The s^2 represents the variance of the calculated values around the mean, while the $s_k^2(x)$ is the variance (or mean square error) of the regression fit, and is given by:

$$s_{k(x)}^2 = \frac{1}{(n-2)}\left[\sum_{i=1,n}(k_i - \bar{k})^2 - \frac{\left[\sum_{i=1,n}(x_i-\bar{x})(k_i-\bar{k})\right]^2}{\sum_{i=1,n}(x_i-\bar{x})^2}\right] \qquad \text{Eq. 3-14}$$

which can also be written as

$$s_{k(x)}^2 = \frac{\sum_{i=1,n}[k_i - k_c(x_i)]^2}{n-2} \qquad \text{Eq. 3-15}$$

The s_w^2 is the within-variance of the data:

$$s_w^2 = \frac{1}{n} \sum_{i=1,n} \sigma_i^2 \qquad \text{Eq. 3-16}$$

The term σ_i in Equation 3-16 is the standard deviation associated with k for a Monte Carlo calculation. Although not required to be included in the calculational uncertainty, the within variance is used to augment the calculational variance. For deterministic codes, which do not have a standard deviation associated with a computed value of k, this standard deviation is zero.

Substituting for $\beta(x)$ in Equation 3-5, an expression for the upper safety limit (for the case when the bias exhibits a trend with one parameter) may be written as:

$$USL(x) = 1.0 - a_0 + bx - \Delta\beta(x) - \Delta k_m, \qquad \text{Eq. 3-17}$$

The a_0 (where $a_0 = 1 - a$) and b parameters are determined from the linear regression, while $\Delta\beta(x)$ is computed from Equation 3-11. The administrative safety margin, Δk_m, is typically assigned a value of 0.05 in safety analyses.

The Equation 3-17 function is represented by the lowermost line of Figure 3-3. As previously discussed, this line represents an upper bound to ensure subcriticality for a given configuration when the calculated k_{eff} plus uncertainty for the configuration is less than the USL. USLs may be calculated for a number of independent parameters for a given system.

Besides providing a statistically valid methodology to establish criticality calculational method bias and bias uncertainty over a defined range of experimental conditions, the lower prediction band technique provides a mechanism to justify extending the range of applicability. As discussed earlier, ANSI/ANS-8.1 allows the range of applicability to be extended beyond this range by extrapolating the trends established for the bias. However, no precise guidelines are specified for the limits of extrapolation.

3.2.3 Calculational Requirements

ANSI/ANS-8.1 requires that a reactivity safety margin be prescribed that is sufficient to ensure subcriticality. The safety margin shall include allowances for uncertainties in the bias and for uncertainties in any extensions of the area(s) of applicability. The proper statistical tools for the calculation of the safety margin (USL) have been presented in the previous section. Specific requirements for actinide-only burnup credit criticality validation have been determined and are stated and discussed below.

All analyses must be performed individually on the two subsets (UO_2, MOX) of critical experiments encompassing the benchmark set of critical experiments. An independent USL must be determined for each subset of critical experiments, and then combined conservatively by selecting the lower of the two at any value of the trending parameter(s). This combination method bounds a USL computed with the entire benchmark set, which may or may not be statistically

valid due to the combination of the two independent subsets. An additional restriction is that the USL must always be lower than 0.95, to eliminate the possibility of using a positive bias.

The range of experimental conditions included in the benchmark experiments selected for validating burnup credit criticality analysis methods establishes the areas of applicability over which the calculated method bias, included in the USL, can be applied. Table 3-2 and Figures 3-1 and 3-2 summarize the extent to which the benchmark experiment set covers the range of anticipated SNF package conditions. As previously stated, package designers must confirm that the UO_2 subset covers the significant criticality control package design features prior to applying the UO_2 subset without modification. It should be noted that although a limited set of experimental data exists for burnup credit analysis method benchmark purposes, the level of extrapolation beyond that typically required for fresh fuel analysis methods is not large. Extrapolations are limited to covering the range of isotopic compositions and neutron spectrum anticipated for systems containing spent nuclear fuel. Benchmark data provided by the MOX critical experiment set provide a strong validation of the analysis method treatment of the key actinide fissile and neutron absorber isotopes. Trending the validation results to a spectral parameter over a diverse group of experiments containing many neutron absorbing materials provides a sound validation basis to address spectrum issues without significant extrapolation. It provides an integral trending basis for many parameters affecting neutron energy spectra, including fissile material content, fuel-to-moderator ratio, and presence of absorbers. The spectral parameter could be the average energy group causing fission (AEG), average lethargy for absorption (ALA), fission (ALF), capture (ALC), or other global spectral parameter.

In addition to trending the calculated multiplication factors against a spectral parameter, trending against initial enrichment (for the UO_2 subset), fuel outside diameter and soluble boron concentration must also be performed. Although those three parameters affect the neutron spectrum, they could also account for other deficiencies. For example, a direct error in the cross section values of U-235 or U-238 in the ENDF/B library would be observed in the initial enrichment trending, while the boron concentration trending accounts similarly for errors in the boron cross section values. The trending against the fuel outside diameter aids in noticing deficiencies in the resonance and cell treatment. Trending against plutonium isotopic concentrations to observe direct errors in the ENDF/B library is not necessary since the most limiting USL is taken from cases with no plutonium and those with plutonium concentrations greater than that seen in SNF. Additional trending parameters could have been suggested (other spectral trending parameters, pellet diameter, moderator to fuel volume ratio), but analyses would be repetitive without adding value to the analyses recommended.

To determine the USL for each subset, trending analyses against the four parameters must be performed. A linear regression fit of the calculated multiplication factors against each of the trending parameters, followed by a slope test on each computed regression slope, shall be performed. For cases when no statistically significant trends are found, Equations 3-4 and 3-10 would be used to compute the USL; if one trend is found, Equations 3-5 and 3-11 would be used. For cases when more than one trend is found, multiple regression analyses[3-20] must be followed.

A 0.05 Δk_m administrative safety margin is typically assigned for spent nuclear fuel package applications using the fresh fuel assumption. This margin is also acceptable for use in burnup credit design applications since the magnitude of uncertainty in calculation bias and the level of extrapolation from experimental conditions are consistent with fresh fuel assumption applications.

3.3 METHODOLOGY DEMONSTRATION WITH SCALE 4.2

A USL to establish definitive bias and uncertainty terms for use in burnup credit analyses is derived using the results summarized in Table 3-1 for the CSAS/27BURNUPLIB system. This example may be used for guidance in performing similar calculations with a different criticality code or code system, or cross section set. For this analysis, the average lethargy for absorption was selected as the spectral trending parameter. Table 3-1 provides ALA values computed from the CSAS/27BURNUPLIB code system outputs.[3-16]

Benchmark calculation k_{eff} results were evaluated against the four trending parameters. Statistical trending analyses were performed on each experiment 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 subset. Results for the trending analyses are presented in Tables 3-3 and 3-4 for the UO_2 and MOX subsets, respectively. The bias for the plutonium-bearing MOX critical experiments are consistent with previously observed trends relating to current plutonium cross section data and are not limited to the CSAS/27BURNUPLIB criticality analysis method.[3-22, 3-23]

Table 3-3. Trending Analyses Results for UO_2-only Subset

Parameter	N	slope	S_{xx}	SS_R	T	$t_{\alpha/2,n-2}$	TREND
ALA	21	4.09E-4	1.06E1	1.66E-4	0.451	2.09	NO
Initial Enrichment	18	9.61E-4	2.41E1	1.30E-4	1.656	2.12	NO
Clad Outside Diameter	18	5.59E-3	3.21E-1	1.42E-4	1.060	2.12	NO
Boron Concentration	21	-6.38E-7	1.69E7	1.61E-4	0.901	2.09	NO

Table 3-4. Trending Analyses Results for MOX Subset

Parameter	N	slope	S_{xx}	SS_R	T	$t_{\alpha/2,n-2}$	TREND
ALA	36	4.55E-3	2.28E1	7.05E-4	4.771	2.03	YES
Clad Outside Diameter	35	-7.93E-3	9.89E-1	9.04E-4	1.506	2.03	NO
Boron Concentration	36	3.41E-6	2.13E6	1.15E-3	0.855	2.03	NO

Using the approach described in the previous section, the various terms required to determine the USL for each subset are computed. Tables 3-5 and 3-6 summarize the intermediate results and the final USL computed for each subset of critical experiments. The benchmark calculation k_{eff} results and USL calculation parameters are plotted in Figure 3-4 for the UO_2 subset and Figure 3-5 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.

Table 3-5. Parameters Used in Upper Safety Limit (USL) Calculations for UO_2-only Subset

Parameter	Value
n	21
Average multiplication factor [$\overline{k}_c$]	0.9938
Average Bias [$\overline{\beta}$]	0.0062
Variance around mean [s^2]	8.380E-6
Within-data variance [s_w^2]	1.719E-6
Pooled standard deviation [s_p]	3.178E-3
$t_{\gamma 1,n-1}$	1.725
Uncertainty [$\Delta\beta$]	5.611E-3
Administrative margin [Δk_m]	0.05
USL	0.9381

Table 3-6. Parameters Used in Upper Safety Limit (USL) Calculations for MOX Subset

Parameter	Value
n	36
Linear regression fit [k (ALA)]	0.9163 + 0.004550 * ALA
Bias [β(ALA)]	0.0837 - 0.004550 * ALA
Average ALA	18.54
Variance of fit [$s^2_{k(x)}$]	2.072E-5
Within-data variance [s^2_w]	1.347E-6
Pooled standard deviation [s_p]	4.698E-3
$t_{\gamma 1,n-2}$	1.691
S_{xx}	22.76
Uncertainty [Δβ(ALA)]	7.944E-3* [1.028 + (ALA-18.54)^2/22.76]$^{1/2}$
Administrative margin [Δk_m]	0.05
USL	0.8663 + 0.004550 * ALA - 0.001665 * [23.40 + (ALA-18.54)^2]$^{1/2}$

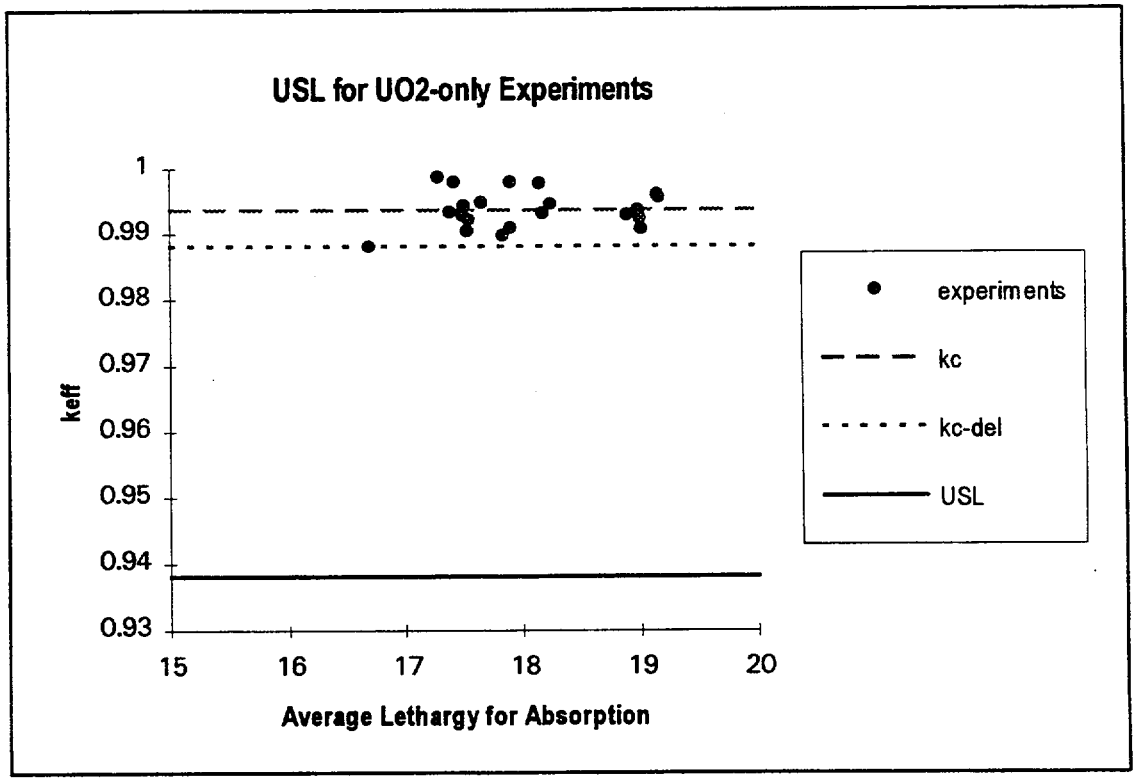


Figure 3-4. Upper Safety Limit (USL) for UO_2 Experiment Subset

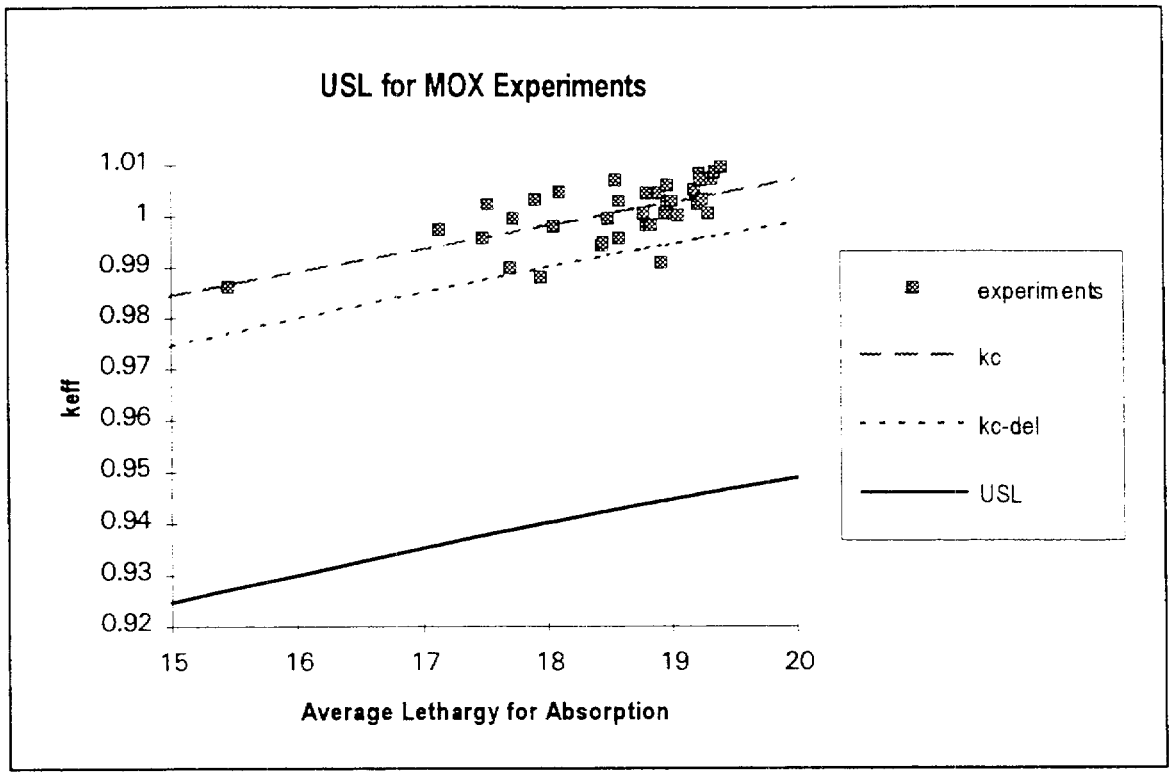


Figure 3-5. Upper Safety Limit (USL) for MOX Experiment Subset

From the combination of the subsets' USLs, the final USL is

$ALA \leq 17.58$: $USL = 0.8663 + 0.004550*ALA - 0.001665*SQRT[23.40 + (ALA-18.54)^2]$

$17.58 \leq ALA$: $USL = 0.9381$

The value of 17.58 corresponds to the intersection of the two USLs. Figure 3-6 presents the final, combined USL.

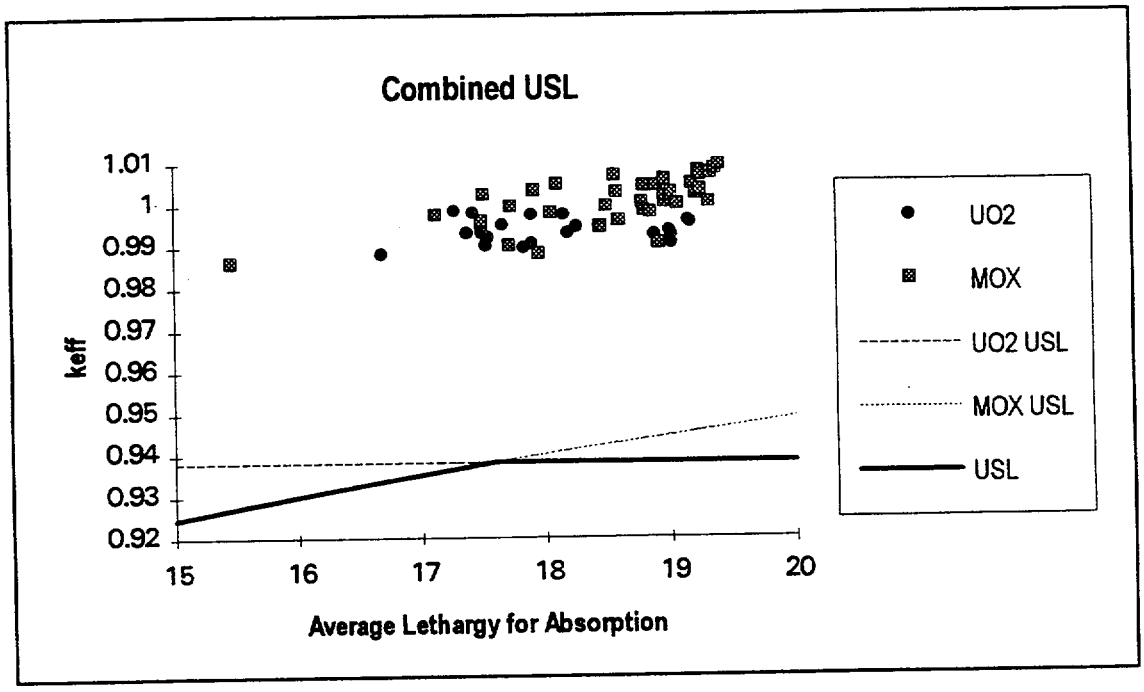


Figure 3-6. Final Upper Safety Limit (USL)

3.4 SUMMARY AND CONCLUSION

This chapter presents a set of critical experiments for validation of criticality calculations and a method for using these experiments to determine a USL on k_{eff}, and it demonstrates the use of the data and method. The 57 experiments selected have been shown to cover the range of expected conditions for the SNF package. Although they do not include actual spent fuel, they include MOX fuel to cover Pu isotopes. The method uses the data to establish a USL on k_{eff}. This is accomplished by calculating a bias and the uncertainty on that bias as a function of trending parameters. Depending on the trending analyses results, the uncertainty is subtracted from either the best fit on k_{eff} or the calculated mean value for the critical experiments. An additional conservative factor, the 0.05 administrative safety margin, is then subtracted resulting in the USL. The use of the data and method is demonstrated with the CSAS/27BURNUPLIB code system. For that analysis, the average lethargy for absorption was selected as the spectral trending parameter, and a significant trend was found for the MOX subset only.

The specific calculational requirements are discussed within the methodology. The following must be performed: 1) analysis of the 57 critical experiments with code system to be validated, 2) regression analyses for each critical experiment subset independently, against a spectral parameter, initial enrichment, outside clad diameter, and soluble boron concentration, 3) a trend test on the linear regression slope for each trended parameter, to determine if the observed trend against the given parameter is statistically significant, 4) determination of a USL, as a function of the

significant trends, for each subset, and 5) combination of the two USLs conservatively by taking the lowest of the two at any value of the trending parameter(s).

This topical report specifically seeks NRC acceptance of: 1) the selection of the 57 critical experiments for actinide-only burnup credit analysis, 2) the selected trend analyses requirements, 3) the method of determining the upper safety limit, and 4) the use of the CSAS/27BURNUPLIB code system with a 0.05 Δk_m administrative safety margin to perform actinide-only burnup credit criticality safety calculations in SNF package design.

4. ANALYSIS AND MODELING PARAMETERS

This chapter provides limiting values of parameters that are used in burnup credit analyses. SNF modeling parameters important to depletion and/or criticality calculations are described. Sensitivity studies and assumptions that were made to obtain the limiting values are presented. Some parameters are generic to all SNF assembly design types; consequently a single value of a parameter is used for all PWR fuel types, while others are specific to the fuel assembly design. Parameters that are needed for isotopic calculations are generic to all SNF package system designs, while parameters that are needed in criticality calculations are SNF package system design specific.

The parameters considered are discussed below, in connection with the nuclear phenomena that are affected. The sensitivity studies provide the values that are needed for the burnup credit analysis. The studies cover the range of parameter values appropriate for burnup credit analyses.

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 affect 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, and the fuel pellet temperature.

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 affects the actinide concentration rather indirectly. 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 dependency of reactivity on 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 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 cool 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

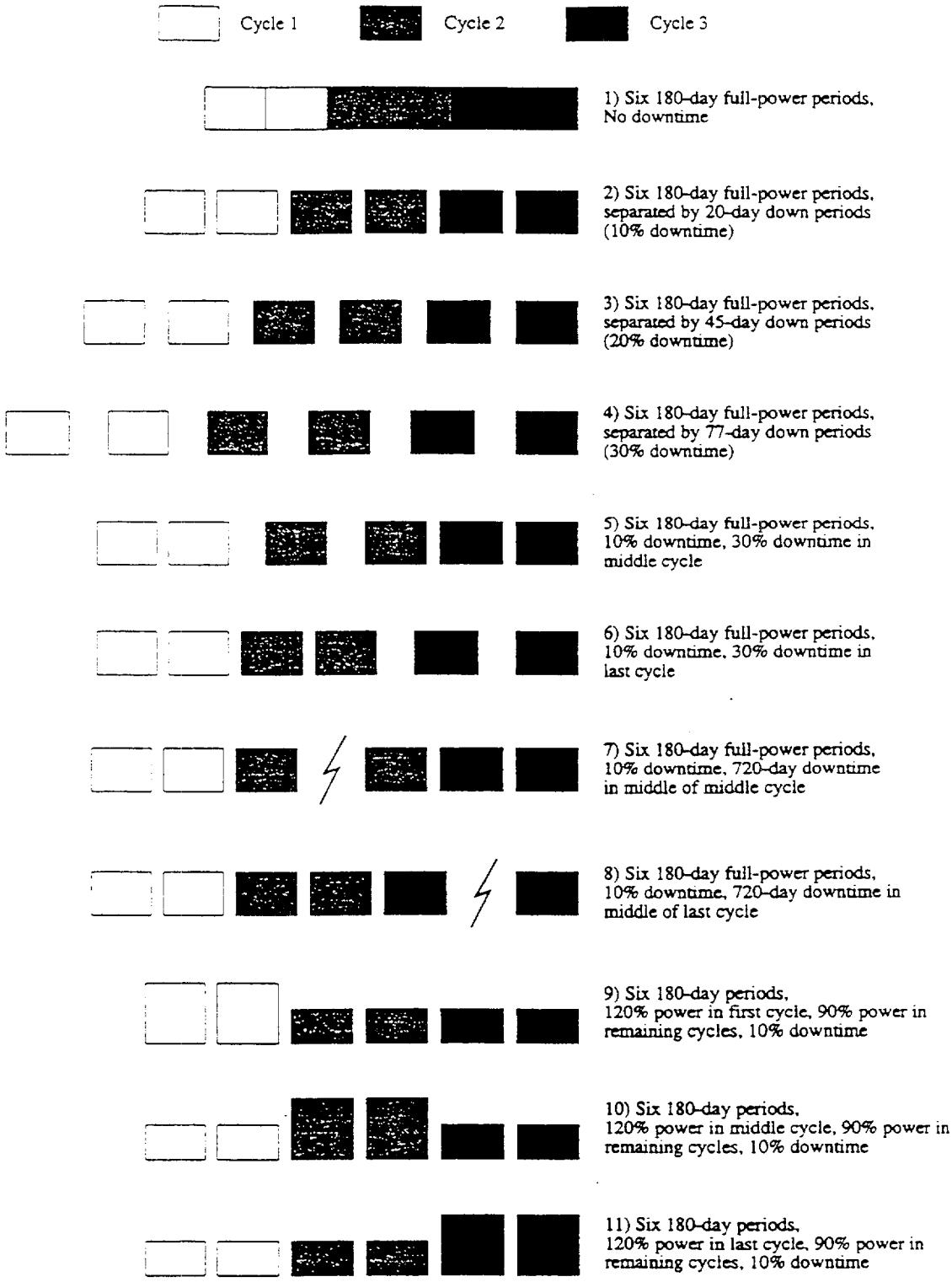


Figure 4-1. Cycle Operating History Cases

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.

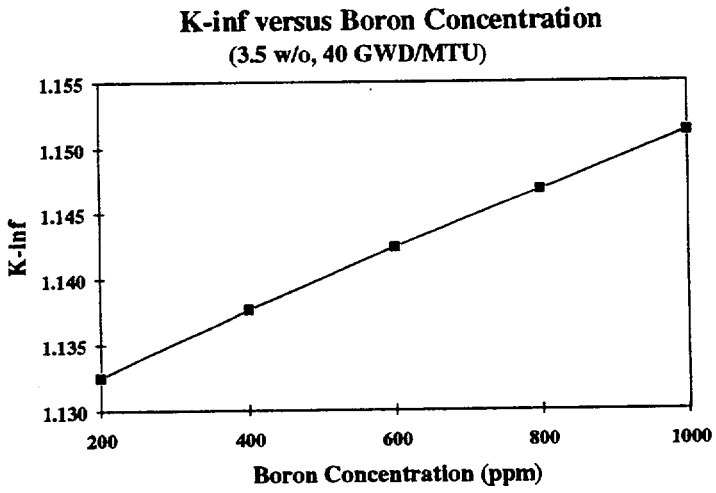


Figure 4-2. k_{inf} versus Boron Concentration

In practice, the boron concentration is adjusted continuously as the fuel is burned, and the critical boron letdown curve is generated as part of the 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 for 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.

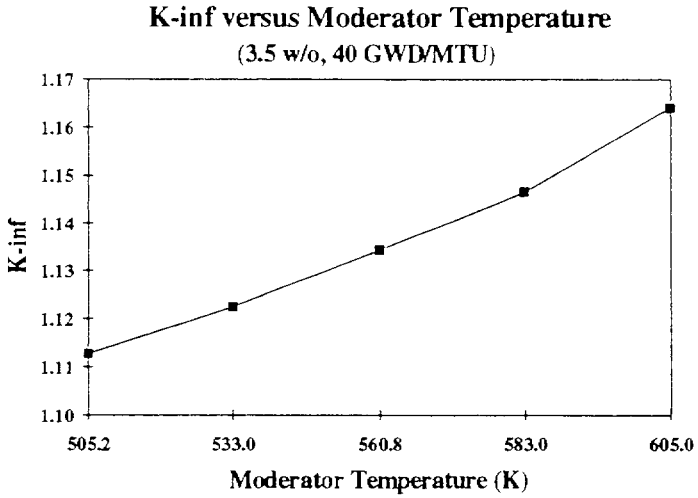


Figure 4-3. k_{inf} versus Moderator Temperature

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 for all 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]

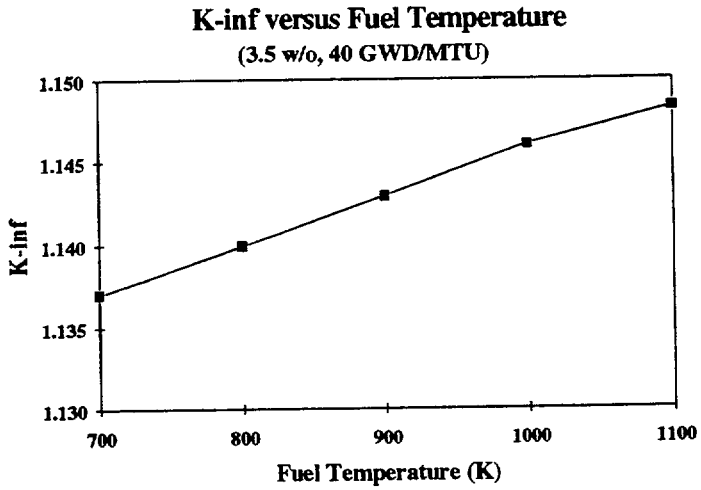


Figure 4-4. k_{inf} versus Fuel Temperature

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 for all assemblies for which the loading curve applies. The maximum average pellet temperature shall be used for burnup credit analysis.

4.1.5 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.

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 0.1 g/cc. Spent fuel baskets in borated water achieve a maximum reactivity at 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. 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 regions. 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 regions determines the sign and magnitude of the end effect. The parameters that influence the end effect include axial burnup profile, axial reflector, cask configuration, fuel assembly length, and cooling time.

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. Restrictions are placed upon the selection of candidate fuel assemblies in Section 6 to ensure that the profiles of assemblies loaded into an SNF package system with burnup credit do not differ significantly from the profile used as a basis for studies in this topical report.

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]

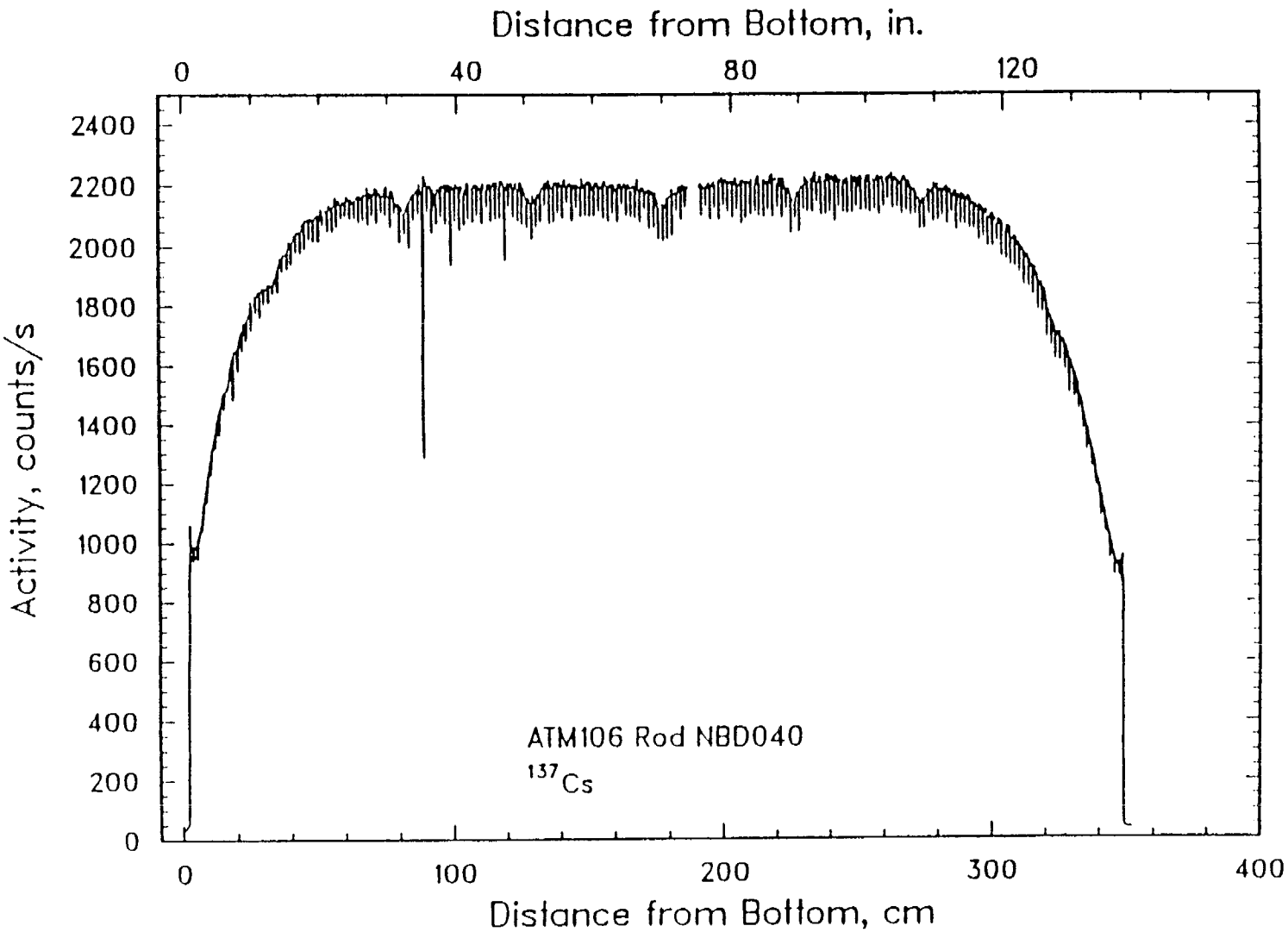


Figure 4-5. Burnup Profile Measurement by Gamma Scan

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

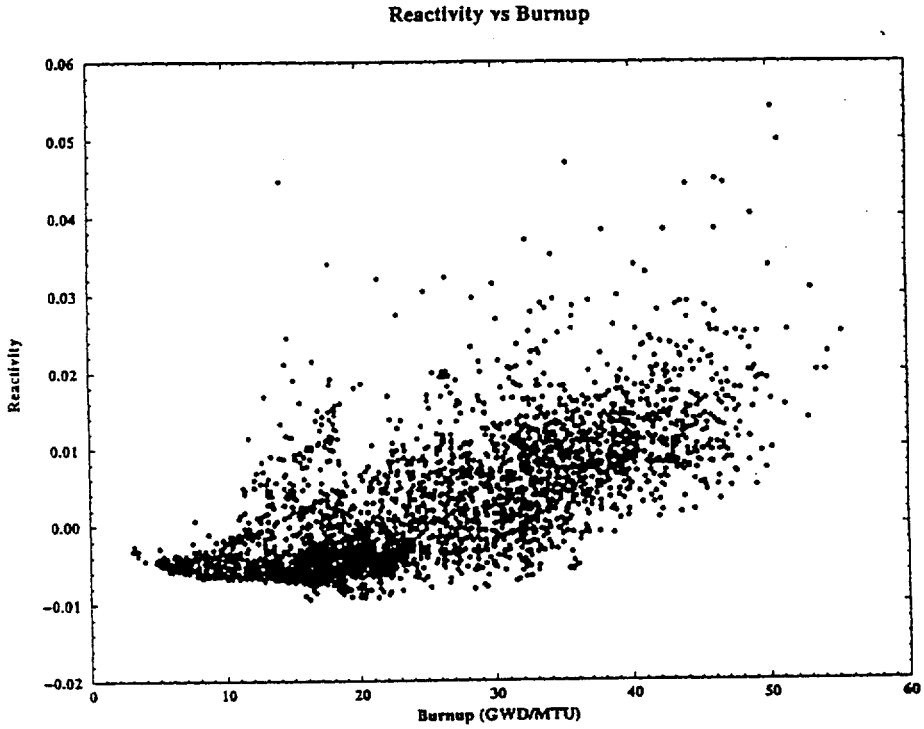


Figure 4-6. End Effect Reactivity versus Burnup

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 profile 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, if a cask has an axially-varying poison plate design, the minimum poison concentration is to be assumed for the whole length. 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 Simplified Axial Modeling

If cask designers feel that their cask design has a large reactivity margin and want to avoid the time-consuming axially burnup-dependent analysis, a simplified axial modeling approach is available. Instead of 18-node, burnup-dependent analysis, a fuel assembly can be analyzed with an axially uniform burnup at the assembly average burnup value. To account for the end effect, k_{eff} biases presented later in this section must be added to the axially uniform calculation. Because the k_{eff} biases need to encompass every commercial and conceptual cask design, they are determined based on a significantly conservative cask configuration, namely a single assembly configuration. For this reason, the simplified axial modeling approach is recommended only for cask designs with large reactivity margins. The requirements for simplified axial modeling are the same as those given in Section 4.2.3.2 except that a uniform analysis needs to be performed and k_{eff} bias is to be added to the resulting multiplication value.

The following four sections describe the end effect trends with respect to axial reflector, cask configuration, fuel assembly length and cooling time. The k_{eff} bias curves to be used in a simplified axial modeling approach are established based on the observed end effect trends.

4.2.3.3.1 Axial Reflector

The cask designers would use the actual axial characteristics of the fuel and cask; however, for the k_{eff} bias curves, a limiting axial reflector must be determined. Two different axial reflector modeling assumptions, pure water reflector and 50/50 (by volume) homogenous mixture of stainless steel and water, are studied for their effects on the magnitude of the end effect.[4-2] The 50/50 stainless steel and water mixture axial reflector approximates the presence of the top and bottom hardware at assembly end regions. It is shown[4-2] that the pure water reflector assumption is more conservative at a high burnup, while there is no significant difference between the two assumptions at a low burnup. At a low burnup, axial modeling assumptions do not affect the magnitude of the end effect significantly because the axial neutron leakage is minimal. At a high burnup, however, the neutron flux becomes peaked in the fuel end regions and different axial reflector assumptions do change the magnitude of the end effect.

4.2.3.3.2 Cask Configuration

Again, the cask designers will fully model the actual cask configuration; however, for the k_{eff} bias curves, a limiting cask configuration must be determined. The magnitude of the end effect is a function of cask size and poison plates in casks.[4-2] Three cask configurations considered in a sensitivity study[4-2] include: (1) an infinite array to approximate a large cask, (2) a four-assembly configuration to approximate a small cask and (3) a single, unreflected assembly configuration to approximate the presence of completely "black" poison plates. It is shown that the magnitude of the end effect increases with the decrease in cask size and with the addition of poison plates.[4-2] Thus, a single assembly configuration is the most limiting one for the end effect. The end effects for a single assembly configuration are significantly higher than those for a conceptual cask design with poison plates.[4-3]

4.2.3.3.3 Fuel Assembly Length

A sensitivity analysis on three different assembly fuel lengths, 10, 12 and 14 feet, shows that the magnitude of the end effect increases with the increase in the active fuel length.[4-2] The increase in the active fuel length increases the fuel end regions which are the main cause of the end effect. The decrease in axial neutron leakage with respect to the increase in the under-burned fuel end regions is the reason for this observed trend.

4.2.3.3.4 Cooling Time

Pu-241 and Am-241 are the only isotopes that undergo significant concentration changes during the first 100 years after SNF discharge from reactors. Pu-241 has a half-life of 14.4 years and decays to Am-241. At discharge, the concentration of Pu-241 is higher in the fuel mid-region that the fuel end regions. Consequently, the Pu-241 concentration change with time is greater in the fuel mid-region. In addition, the concentration of Am-241, most of which comes from the decay of Pu-241, increases more in the fuel mid-region. These phenomena lead to an increase in relative reactivity of the fuel end regions compared to the fuel mid-region and result in an increased end effect. A sensitivity analysis shows that the increase in the end effect, from 5 to 15-year cooled assemblies, can be as large as 1.0 % in k_{eff} at a high burnup for a single assembly configuration.[4-2]

4.2.3.3.5 k_{eff} Bias Curves

The k_{eff} bias curves to be applied in the simplified axial modeling approach are determined based on the single assembly cask configuration, the pure water axial reflector, and the limiting axial burnup profiles given in Table 4-3. Separate curves are provided for different fuel assembly lengths and cooling times. Figures 4-7 through 4-9 show the k_{eff} bias curves corresponding to 5, 10 and 15-year cooling time.[4-2] The k_{eff} bias curves are piecewise straight lines in three burnup zones consistent with the limiting axial profiles shown in Table 4-3. The k_{eff} bias values at different burnups are shown in Tables 4-4 through 4-6.[4-2] The end effect is defined as Δk_{eff} (%) = $100*(k_{18\ nodes} - k_{uniform})$ in Figures 4-7 through 4-9 and Tables 4-4 through 4-6.

Any cask designed for a cooling time greater than 15 years cannot use the k_{eff} bias curves and must resort to axially burnup-dependent analyses. The k_{eff} bias curves can be interpolated between different assembly lengths but not between cooling time. A conservative cooling time must be adopted if an intermediate cooling time not shown in Figures 4-7 through 4-9 is to be used. For example, 7-year cooled, 12.5-foot assembly must use 10-year k_{eff} bias curves for interpolation between 12 and 14-foot values.

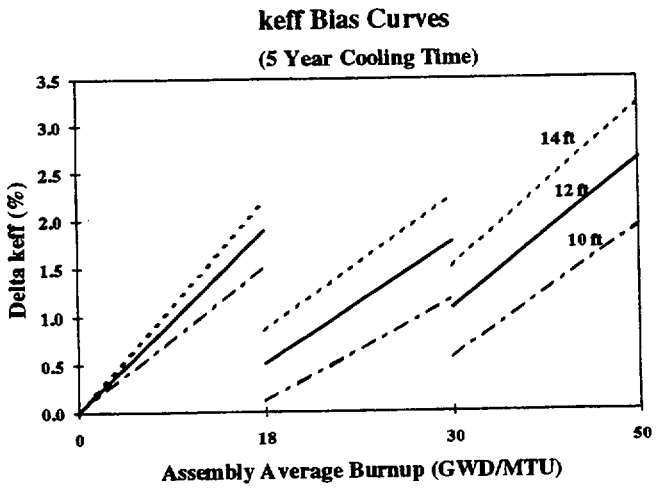


Figure 4-7. k_{eff} Bias Curves for 5-Year Cooling Time

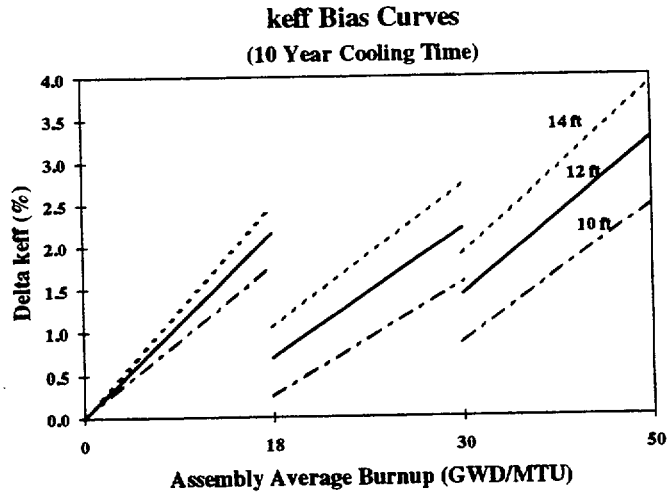


Figure 4-8. k_{eff} Bias Curves for 10-Year Cooling Time

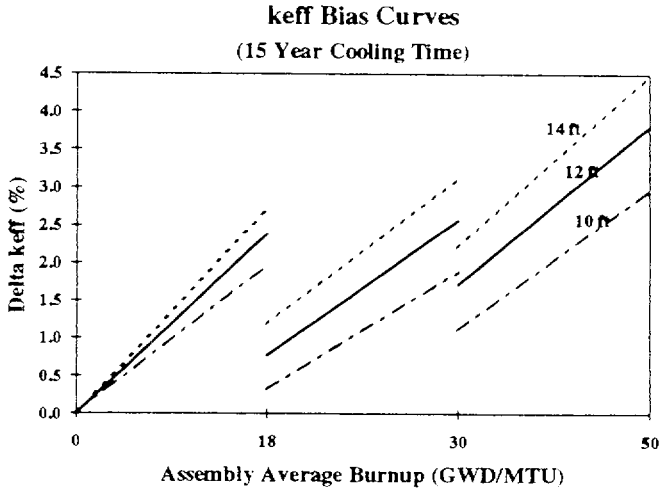


Figure 4-9. k_{eff} Bias Curves for 15-Year Cooling Time

Table 4-4. k_{eff} Bias Values for 5-Year Cooled Assembly

Average Burnup (GWD/MTU)	Δk_{eff} (%)		
	10-foot Assembly	12-foot Assembly	14-foot Assembly
0	0	0	0
18-	1.51	1.89	2.18
18+	0.11	0.50	0.85
30-	1.17	1.76	2.24
30+	0.56	1.07	1.50
50	1.92	2.63	3.22

Table 4-5. k_{eff} Bias Values for 10-Year Cooled Assembly

Average Burnup (GWD/MTU)	Δk_{eff} (%)		
	10-foot Assembly	12-foot Assembly	14-foot Assembly
0	0	0	0
18-	1.77	2.16	2.46
18+	0.25	0.69	1.05
30-	1.57	2.20	2.73
30+	0.86	1.42	1.89
50	2.48	3.26	3.91

Table 4-6. k_{eff} Bias Values for 15-Year Cooled Assembly

Average Burnup (GWD/MTU)	Δk_{eff} (%)		
	10-foot Assembly	12-foot Assembly	14-foot Assembly
0	0	0	0
18-	1.97	2.38	2.69
18+	0.32	0.76	1.19
30-	1.89	2.57	3.12
30+	1.11	1.71	2.21
50	2.97	3.81	4.49

4.2.3.4 Summary of Axial Burnup Profile

Axially burnup-dependent analyses and k_{eff} bias curves 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. Further, if a fuel assembly employs multiple axial enrichment design, the maximum enrichment is to be assumed for the entire assembly. 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 theviewpoint of the end effect. 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. There are no initial enrichment or burnup limits in using axially burnup-dependent analyses from the viewpoint of the end effect. However, the burnup limit is 50 GWD/MTU if k_{eff} bias curves are used. The k_{eff} bias curves are not intended to be extrapolated beyond the ranges shown in Figures 4-7 through 4-9.

4.2.4 Horizontal Burnup Profile

A significant horizontal variation in burnup can exist in individual PWR assemblies particularly if they are irradiated near the periphery of a core and 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 unacceptably high reactivity 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 4-10 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 4-10 that the values given in Table 4-7 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 4-7. 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.

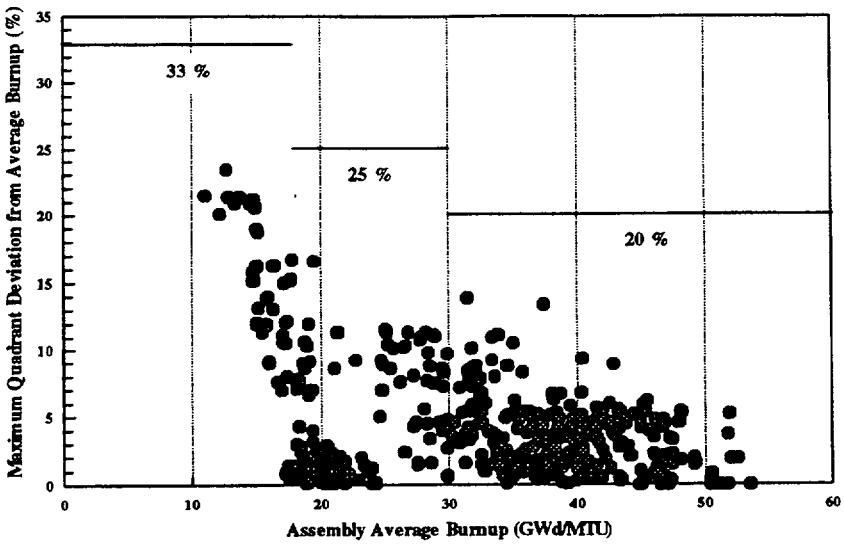


Figure 4-10. Maximum Quadrant Deviation versus Assembly Average Burnup[4-6]

Table 4-7. 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 4-8 below.

Table 4-8. 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 (293K)
Axial Burnup Profile	Criticality	Axially Burnup-Dependent, 18-Node Analysis with Profiles in Table 4-3 or Uniform Analysis with k_{eff} Bias Curves
Horizontal Burnup Gradient	Criticality	Horizontal Burnup Gradients in Table 4-7

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 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 5-2. 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.

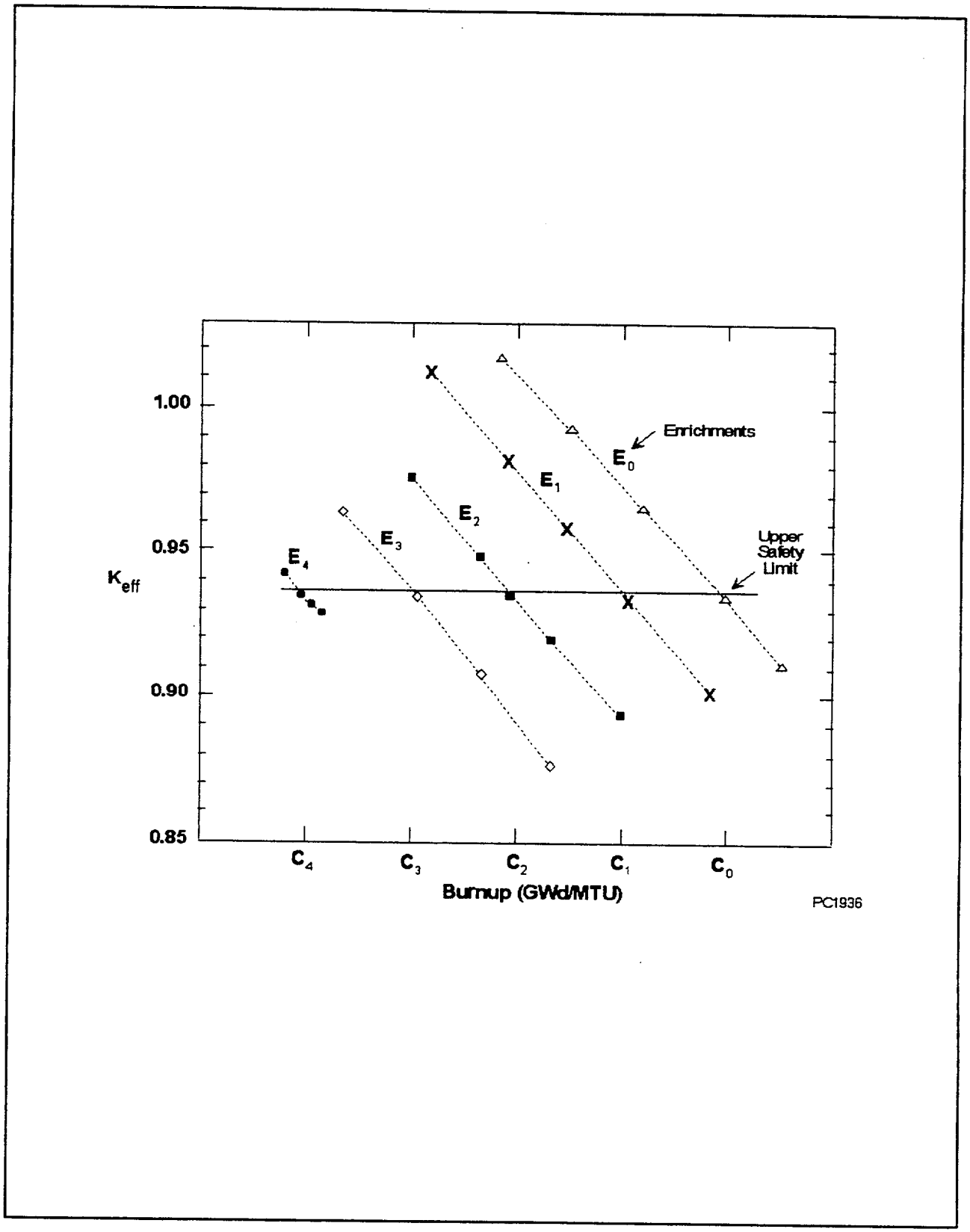


Figure 5-1. Determination of Required Minimum Burnups for a Specific SNF Package

5.2.2 Find the Limiting Intial 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 intial enrichments result depending on the axial bunrup profiles (or k_{eff} bias values) and horizontal burnup gradient selected at 18 and 30 GWd/MTU. It is required that the minimum of the two minimum intial enrichments be determined. This can be achieved by adopting axial burnup profiles 1 and 2 in Table 4-3 (or higher k_{eff} bias values) and the horizotal burnup gradients of 33% and 25% for 18 and 30 GWd/MTU, respectively. Deterimining the other initial enrichment is not required. However, it can be achieved by adopting axial burnup profiles 2 and 3 in Table 4-3 (or lower k_{eff} bias values) and the horizotal bunrup 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.

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). The required minimum intial 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.

A spent fuel assembly that has a verified 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.

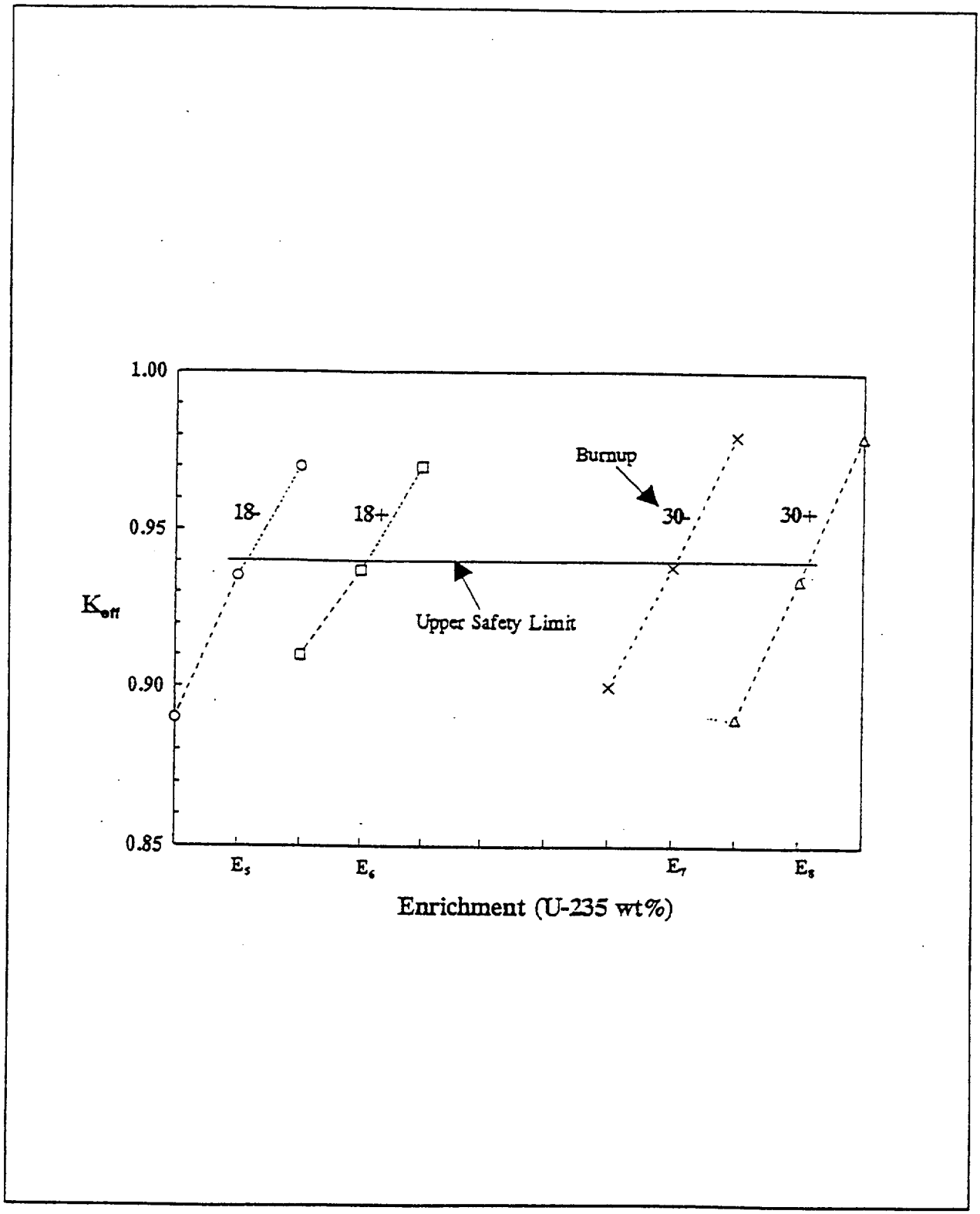


Figure 5-2. Determination of Limiting Intial Enrichment at Burnup Discontinuities

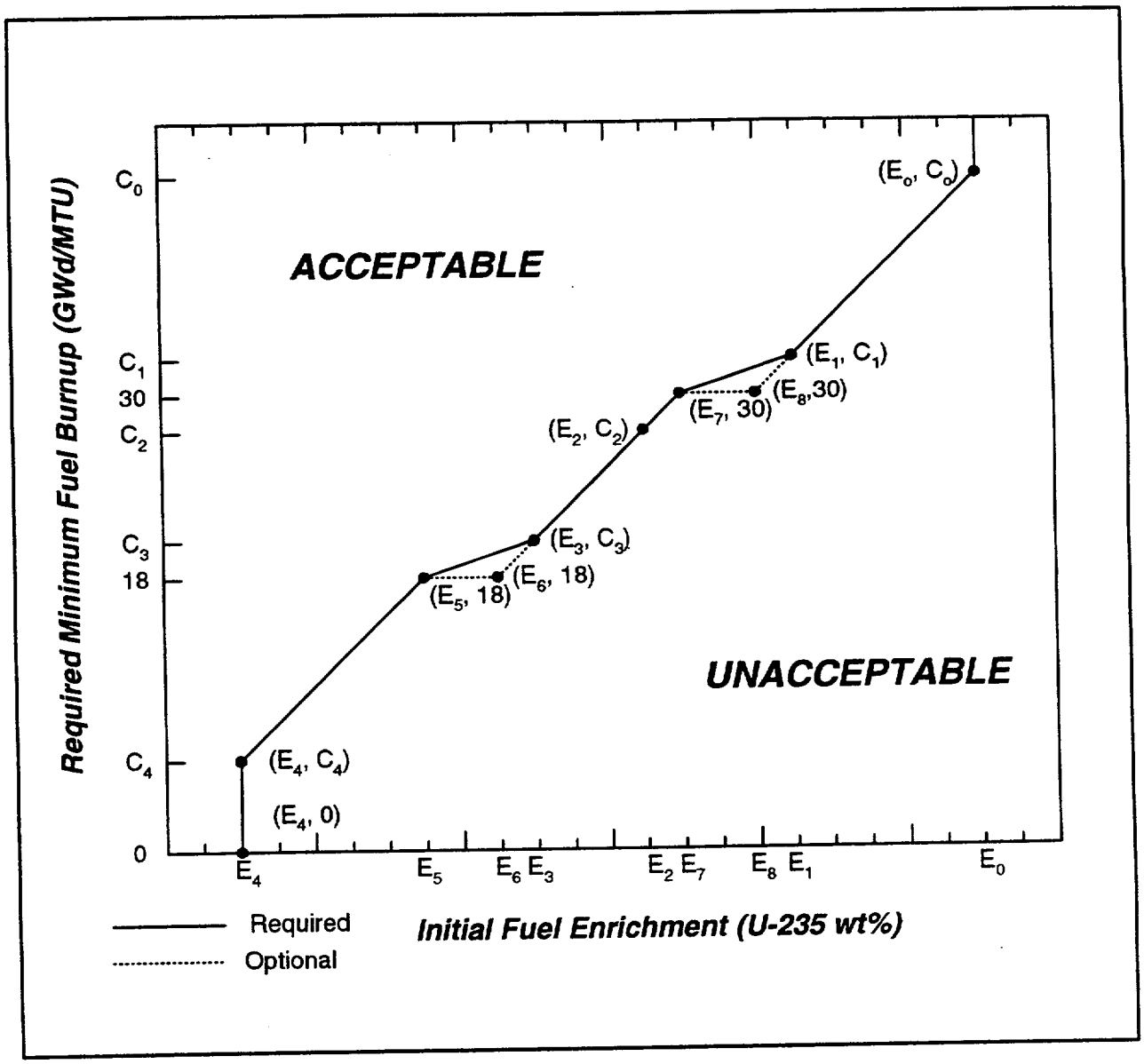


Figure 5-3. Development of the Burnup Credit Loading Curve for a Specific SNF Package

5.2.4 Burnup Records Adjustment

The loading curve is based on the assured minimum assembly-average burnup. Therefore, the loading curve burnup value should be compared to the utility-maintained reactor records for that assembly, reduced for the uncertainty. Thus, it is important to quantify the uncertainty associated with each assembly's reactor record burnup, and to make the proper adjustment before comparing to the loading curve. It is difficult to establish a single reactor record uncertainty applicable to every assembly, regardless of age, reactor plant type, and calculational method. Therefore, each utility determines the appropriate nominal reactor record assembly average burnup for the assemblies they intend to load into burnup credit casks, and determines the associated burnup uncertainties via utility-developed, approved procedures. Note that in deriving reactor record burnup uncertainties, conservative bounding values (e.g., derived from the plant's Final Safety Analysis Report) may be used where appropriate for the standard deviation, effective degrees of freedom, or any of the parameters discussed in the guidelines below. Guidelines for development of utility-specific procedures to determine reactor record burnup and burnup uncertainties are as follows:

1. Use the in-core detector system to establish relative assembly-average power (RelAssm), with an uncertainty consistent with power distribution uncertainty.

2. Establish core power (P) using calibrated instrumentation with known uncertainties (e.g., flowmeters, thermocouples, etc.).

3. Determine the nominal reactor record assembly-average burnup ($RRAA_{Burnup}$) from:

$$RRAA_{Burnup}(\text{MWD/Assy-MTU}) = \left[\int_{\text{assem. lifetime}} [RelAssm(t)*P(t)]dt\right] / MTU_{assem}, \quad \text{Eq. 5-1}$$

or

$$RRAA_{Burnup}(\text{MWD/Assy-MTU}) = \text{sum}_i\ [RelAssm_i(t) * P_i(t)*\Delta\ t_i]/MTU_{assem} \quad \text{Eq. 5-2}$$

4. Document the nominal reactor record assembly-average burnup ($RRAA_{Burnup}$) for each assembly to be loaded into a burnup credit cask/canister, and the specific method by which periodic observations of RelAssm and P are used to establish RelAssm(t) and P(t). Also document the uncertainties associated with the input data (RelAssm(t), P(t), and MTU_{assem}) and the calculated nominal reactor record assembly-average burnup ($RRAA_{Burnup}$) from step 3. The uncertainty of the calculated nominal reactor record assembly-average burnup ($RRAA_{Burnup}$) should be based on the statistical propagation of errors in the terms in equation 5-2.

5. The minimum assembly-average burnup ($MinAA_{Burnup}$) shall be used for loading a cask for burnup credit. Hence, for cask loading, the minimum assembly average burnup, defined as:

$$MinAA_{Burnup} < RRAA_{Burnup} - 1.645\ \sigma_{RRAA} \quad \text{Eq. 5-3}$$

must be greater than the loading curve value at the assembly's initial enrichment.

Since each utility may have assemblies for which burnup has been calculated using different methods, codes, etc., different assemblies may have different reactor record burnup uncertainties. The reactor record uncertainty value associated with each assembly needs to be consistent with, or conservative for, the particular reactor record calculational method used.

Note that there are compelling reasons to believe that most reactor records have uncertainties less than approximately 5%. Among them are the following:

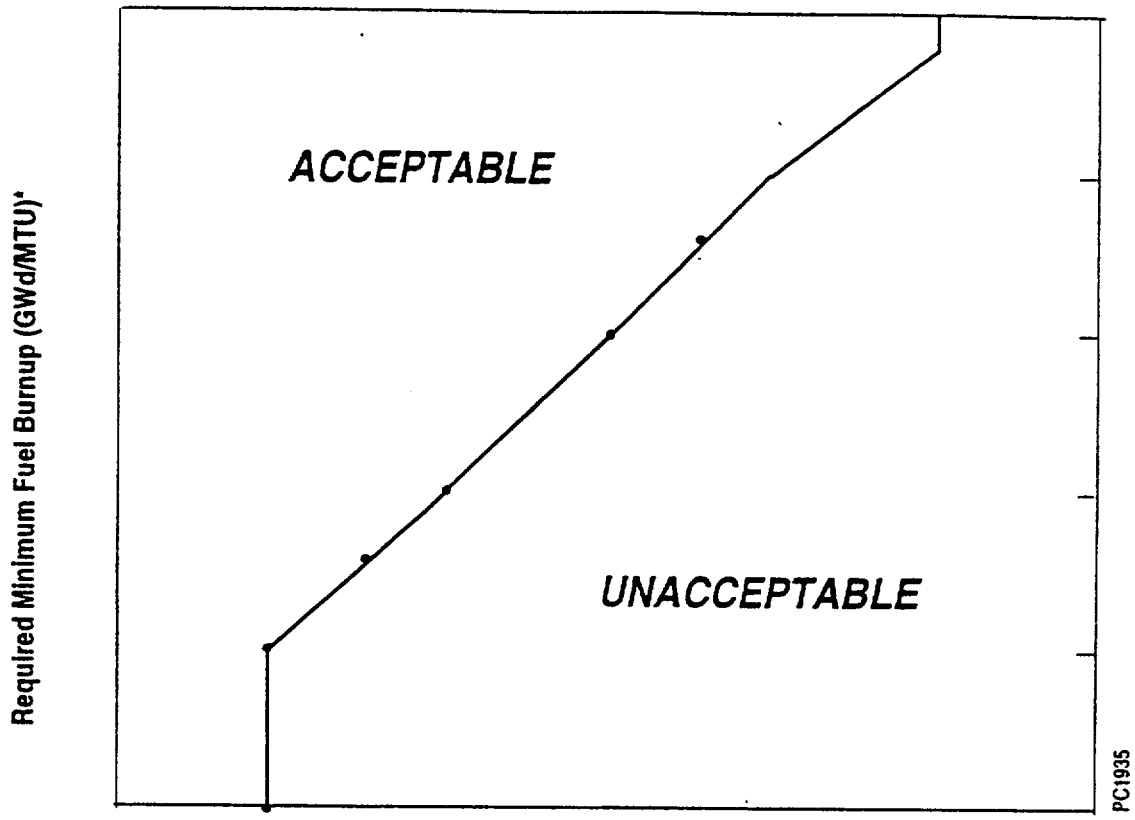
(1) Westinghouse Standard Technical Specifications[5-1] requires that the measured radial peaking factor, $F_{\Delta H}$, be increased by 4% to account for measurement uncertainties, and that measured heat flux hot channel factors be increased by 5%. Both of these uncertainties are for localized, pin-wise power measurements within an assembly. Therefore, the measurement uncertainties associated with the entire assembly (from which assembly-average burnup values are derived) would be expected to be smaller than 4-5%, due to the effects of averaging.

(2) The Westinghouse Standard Technical Specifications[5-1] require daily adjustment of excore neutron detectors to ensure the core power level determined by the excore detectors is within ~ 2% of the value calculated via calorimetry. A monthly calibration of incore detectors using the excore detectors (generally to within 3%) is also required. Since the above limits are technical specification limits, actual variations are likely to be significantly less. Combining these two uncertainties yields the conclusion that incore detector systems are routinely demonstrated to measure core power level to well within ~ 5%. Since incore detector measurements are generally used in assembly burnup calculations, a value of ~ 5% for burnup uncertainty is consistent with the above discussion.

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. For any SNF package design, several loading curves may be generated. Separate loading curves may be generated for each assembly design type, cooling time, and number of removable burnable absorbers.

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.



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

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 should 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 amounts of burnable absorbers in the assembly. This effect is generally small but may be as large as a few percent in reactivity.[5-2]

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 should state whether it applies to fuel with burnable absorbers. Typically, burnable absorber assemblies are removed after one cycle. However, the SNF depletion analysis should 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 should 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.[5-3] Figure 5-5 shows that shortly after discharge from the reactor, the reactivity decreases monotonically for the first 100 years. 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. The negative reactivity worth of fission products increases with cooling time; therefore, neglecting fission products adds more conservatism with cooling time.

Since additional cooling time decreases reactivity during the first 100 years, a loading curve would be valid for any cooling time greater than that used in the analysis. After 100 years of cooling, the reactivity starts to increase due to Pu-240 decay. This topical report does not analyze or provide parameter limits to cover this increase in reactivity; therefore, the scope of this topical report is limited to 100 years of cooling time. The cooling time used in the analysis must be placed on the loading curve. The records verification presented in the next chapter is used to verify that the cooling time is greater than the value shown on the loading curve.

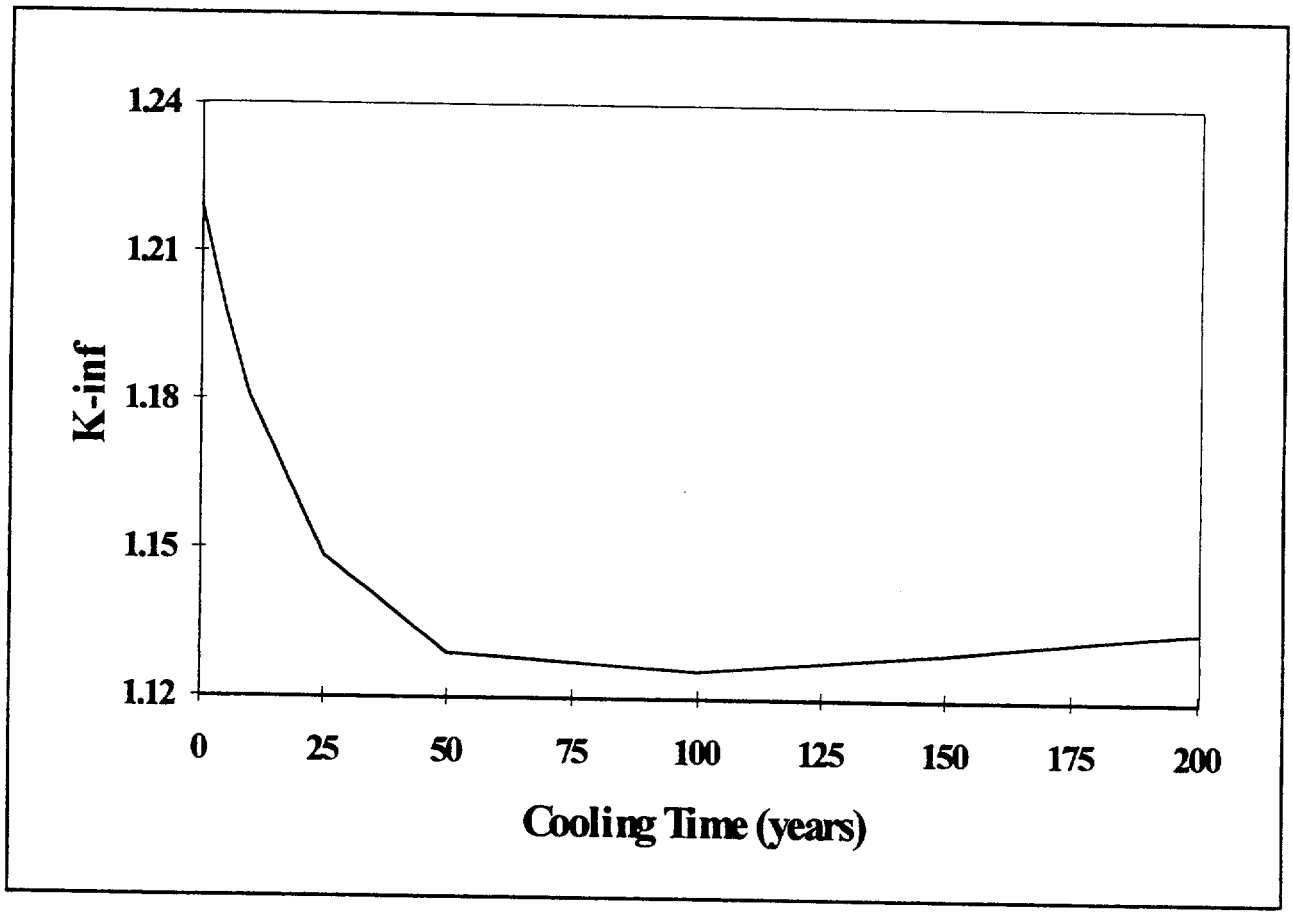


Figure 5-5. k_{inf} versus Cooling Time (Actinides only, 3.0 wt. % U-235, 30 GWd/MTU)[5-3]

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 reactor record burnup uncertainty is accounted for by utilities following established guidelines.

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 100 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 by the method described in the next chapter.

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6. PHYSICAL IMPLEMENTATION AND CONTROLS

This chapter addresses the physical implementation and administrative controls that should be imposed in loading a burnup credit package. As discussed previously, burnup credit loading curves specify the criticality control fuel acceptance criteria and serve as operational limits for selecting fuel assemblies for loading into a burnup credit package. The applicable burnup credit loading curve is used along with reactor records, fuel assembly classification and package loading procedures, and an independent burnup verification measurement to ensure that spent fuel assemblies have experienced sufficient burnup to satisfy minimum criticality safety design requirements prior to loading in a burnup credit package. Although specific interfaces with 10 CFR 50[6-1] site operations regulations are not addressed within the scope of this topical report, administrative controls and procedures proposed to ensure proper loading of burnup credit packages are consistent with Regulatory Guide 1.13[6-2] guidance.

A competitive commercial market is developing for burnup measurement systems. This chapter describes the requirements for burnup measurement systems used to verify fuel assembly burnup as a prerequisite for loading into a burnup credit package. Two examples of the technology are found in Appendix B. Both of these systems have been recently tested in the U.S. on PWR fuel assemblies in commercial reactor spent fuel pools, and both have the potential to meet the requirements of this chapter.

6.1 BURNUP CREDIT PACKAGE LOADING PROCESS

Before burnup credit package loading operations, specific burnup credit package loading licensing limits are established. These 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. Confirmation of fuel assembly acceptance status is also required for a number of other fuel design and operating history characteristics.

Burnup credit loading curves specify the criticality control fuel acceptance criteria and serve as the operational limits for selecting fuel assemblies for loading into a burnup credit package. Physical implementation of burnup credit involves facility preparations, including development and implementation of fuel classification procedures and procedures for actual fuel loading operations. A block diagram illustrating the process and procedures involved in the burnup credit package loading process is provided in Figure 6-1. The shaded blocks highlight the items that are unique to the loading of burnup credit packages. The unshaded items are activities that are also associated with standard fresh fuel assumption package loading activities. The following sections discuss the key elements of the loading process illustrated in Figure 6-1.

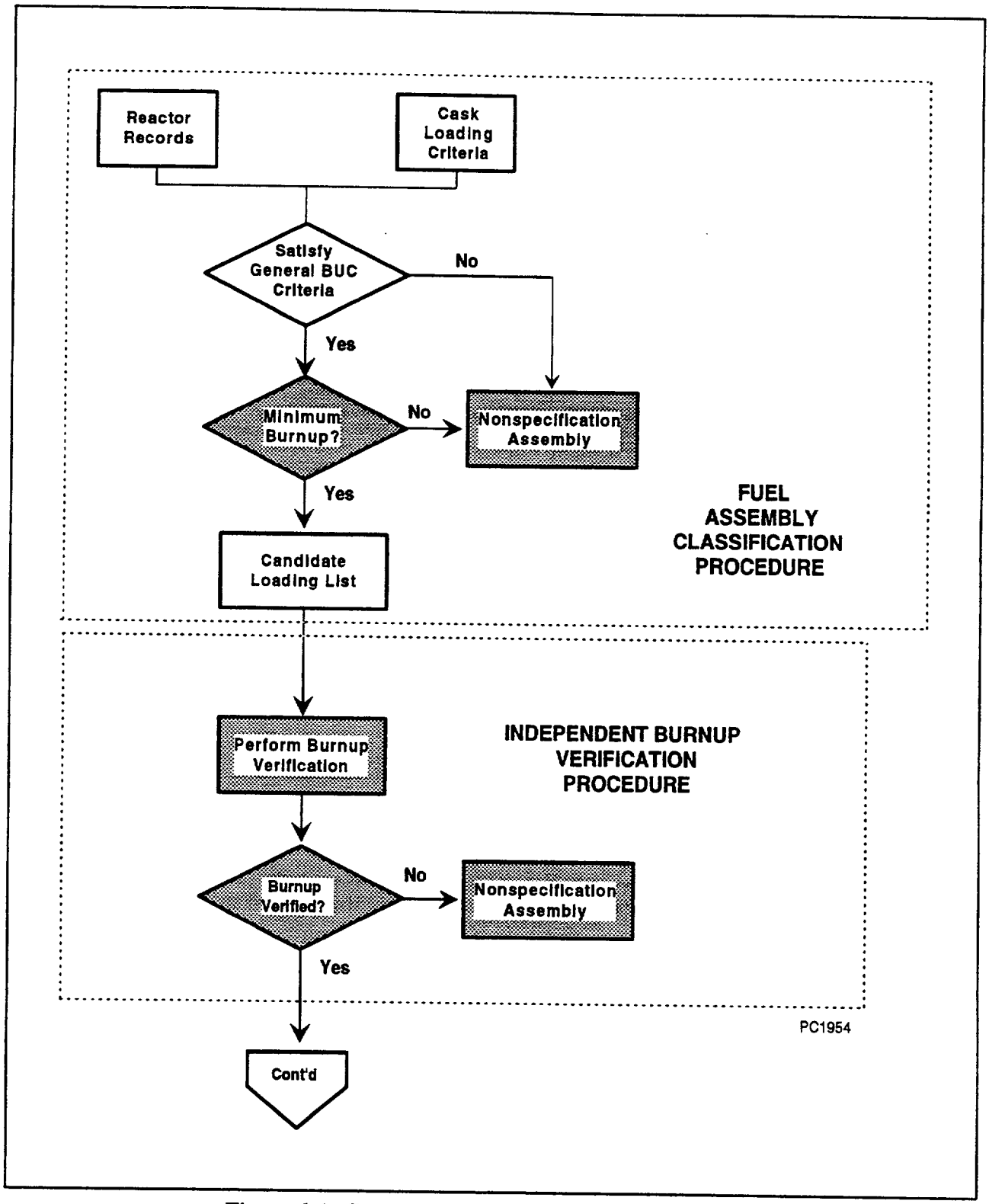

Figure 6-1. Burnup Credit Package Loading Process

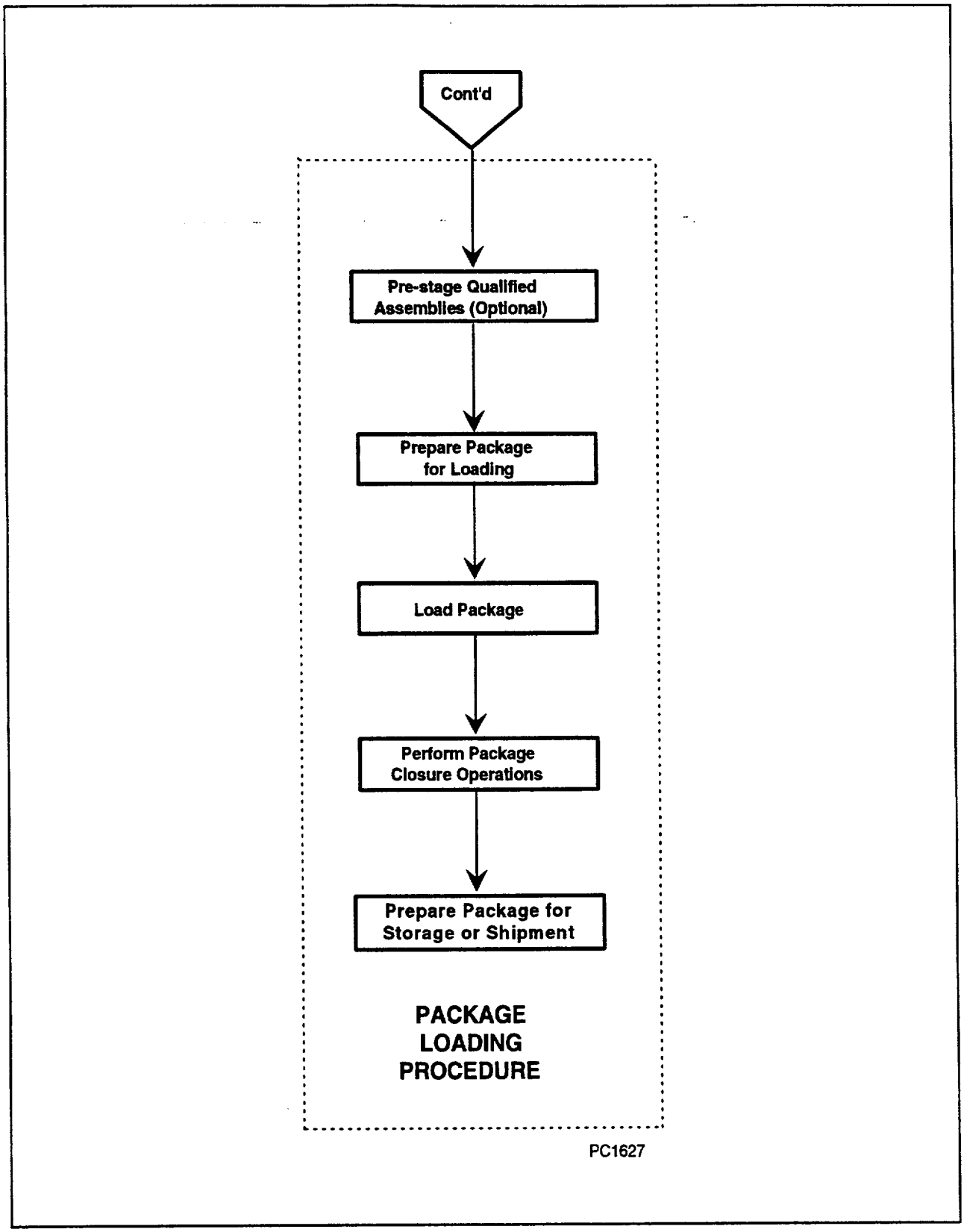

Figure 6-1. Burnup Credit Package Loading Process (Continued)

6.1.1 Fuel Assembly Classification Procedure

Reactor records and the burnup credit package loading criteria are used to classify spent fuel assemblies as acceptable or not acceptable for loading into a burnup credit package. Reactor records are maintained for each assembly received at a reactor site. Records are tracked by a unique alphanumeric assembly identifier physically stamped on each assembly. Reactor record sources include facility-specific Special Nuclear Material accountability records, reactor core design reports, reactor core operating reports, and spent fuel pool inventory records. Records maintained include as-received fuel characteristics such as initial enrichment and physical configuration, current and historical assembly location(s) onsite, and the complete operating history of each assembly while in the reactor core. Operating history parameters recorded include assembly average burnup (calculated based on guidance in Section 5.2.4), average power level, axial power distribution, and non-fuel core power shaping component histories. Physical modifications performed on fuel assemblies are also documented for Special Nuclear Material accountability purposes. The package loading criteria are specified in the package Certificate of Compliance or Safety Evaluation Report, and the supporting Safety Analysis Report. These criteria ensure that criticality, thermal, and radiation design and licensing limits of the package are not exceeded.

Fuel assembly classification procedures will be prepared before loading operations and will be unique to specific package designs. Prior to commencement of burnup credit package loading operations, a Fuel Assembly Classification Procedure is implemented to identify any spent fuel assemblies that do not meet the applicable criticality design requirements specified for the burnup credit package. These assemblies are classified as "nonspecification" spent fuel assemblies and are prohibited from further consideration for loading into the package. Fuel assemblies are classified as nonspecification based on: 1) failure to satisfy general burnup credit criteria or 2) noncompliance with the minimum burnup criterion.

General burnup credit criteria identify fuel assembly attributes that must be considered outside the scope, or outside the range of applicability, of this topical report. These criteria are identified in Section 1.3.

After confirming compliance with the general burnup credit criteria identified in Section 1.3, each fuel assembly is screened for compliance with the minimum burnup criterion based on reactor records. Fuel assemblies that meet the minimum burnup criterion for their respective fuel assembly type, initial enrichment, cooling time, and burnable absorber configuration history are then evaluated with respect to the remaining package loading criteria unrelated to criticality safety. The Fuel Assembly Classification Procedure is performed and documented in a site-specific QA calculation format satisfying 10 CFR 50, Appendix B, requirements. The determination that a fuel assembly satisfies the package loading criterion is based upon comparing the information in reactor records or information derived through calculation, such as decay heat, to loading criteria obtained from the package Certificate of Compliance or Safety Evaluation Report, and the Safety Analysis Report. A fuel assembly that meets all applicable loading criteria is considered to be a "candidate" fuel assembly for loading into the specified package design. The Fuel Assembly Classification Procedure maintains up-to-date lists of fuel assemblies classified as candidate or

nonspecification. The lists will include fuel assembly identification and fuel rack location identifiers, and other pertinent information obtained from reactor records such as assembly initial enrichment, average burnup, burnable absorber status, and discharge date.

6.1.2 Independent Burnup Verification Procedure

Prior to burnup credit package loading, candidate fuel assemblies undergo burnup verification. Burnup verification consists of double verification of the fuel assembly identifier and corresponding cell location, and physical measurement. The verification measurement consists of physically measuring the gamma-ray and/or neutron emissions from the spent fuel assembly and determining if the emissions correspond to those expected from an assembly with the burnup, initial enrichment, and cooling time since discharge, specified in the reactor records. The burnup verification is performed with a measurement system which meets the guidelines of Section 6.4. In addition to confirming proper assembly selection, the measurement system verifies the consistency of the data recorded in the reactor records for each assembly prior to loading. Examples of two candidate measurement systems are briefly described in Appendix B.

The need for an Independent Burnup Verification Procedure is unique to the use of burnup credit packages. Criticality control loading restrictions for packages designed using the "fresh fuel assumption" only require confirmation that fuel assemblies satisfy initial enrichment limitations prior to loading. Enrichment confirmation is done purely via administrative controls (e.g., independent checks of assembly numbers prior to loading), whereas for burnup, an actual measurement is performed to augment the administrative controls. If the verification confirms the consistency of the burnup and cooling time values assigned by the Fuel Assembly Classification Procedure, the candidate fuel assembly is classified as qualified for loading into the burnup credit package. Qualified spent fuel assemblies may be moved to a segregated region of the spent fuel pool for eventual package loading. A list of qualified fuel assembly identifiers and corresponding fuel pool location identifiers is maintained, and the reactor records are updated accordingly.

If an inconsistency between the assigned assembly burnup or cooling time and the measurement system value for that assembly is identified, the fuel assembly is classified as nonspecification. Section 6.3 discusses the criteria for determining such inconsistencies. Although the general disposition of nonspecification fuel assemblies is beyond the scope of this topical report, burnup credit package loading procedures will require investigation of inconsistencies detected between reactor records and the measurement system values. The focus of the investigation procedure will be to identify and correct any possible errors in reactor records which could contribute to future verification inconsistencies or loading errors.

6.1.3 Package Loading Procedure

The Package Loading Procedure governs activities related to facility-specific pre-staging of qualified assemblies, double verification of the fuel assembly identification numbers prior to and following package loading, loading the qualified assemblies into the package, and package closure operations.

The need for a Package Loading Procedure is not unique to the use of burnup credit packages. Control of package loading operations is required to ensure that only qualified assemblies are loaded. Prior to fuel movement out of an existing pool storage location, the Package Loading Procedure requires independent double verification of the assembly identifier and corresponding fuel pool location by two fuel handling operators. Following confirmation that the proper assembly is engaged by the fuel handling device, individual fuel assemblies are removed from the fuel rack, moved to the package loading area, and placed into the predesignated fuel cell location in the package. After placement in the package, the fuel assembly identifier and corresponding package fuel cell location is again independently double verified. Upon completion of the package loading, each fuel assembly identifier and package location is again double verified.

6.2 FUEL ASSEMBLY MISLOAD

The burnup credit package loading process described in the previous section provides sufficient control over nuclear criticality safety practices to satisfy the Double Contingency Principle of ANSI/ANS-8.1.[6-3] Assembly identifiers are independently verified by two fuel handling operators at each stage of the burnup verification and package loading procedures, and the reactor records assigned 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.

6.3 BURNUP VERIFICATION REQUIREMENTS

The Independent Burnup Verification Procedure requires a physical measurement to confirm proper reactor records assignment of burnup and enrichment prior to loading specific fuel assemblies into a burnup credit package. A negative result of this measurement is a rejection of the acceptance of the assembly for loading in a burnup credit package. Hence, a rejection criteria must be established. This requirement should be consistent with the need for confirmation as well as the technology available to do the verification. The rejection criterion is that the measured burnup must be within 10% of the reactor record burnup. This is a two-sided requirement since it is desirable to reject any assembly with an unexpected result. Although, measuring a burnup greater than the reactor record by more than 10% may be safe with regard to burnup, it implies a bad record which includes the enrichment. Since no direct measurement of enrichment is required, any indication of an erroneous record must cause a rejection until it is resolved.

The measurement is to confirm the reactor record value of burnup and the uncertainty in this reactor burnup record is accounted for by a related reduction in the burnup before comparing to the loading curve. Disagreement between the measurement and the reactor record is not an uncertainty that needs to be used to reduce burnup credit but rather an indication that something is wrong. The question arises whether an unnoticed error of 10% would lead to an unsafe

condition. First, it is projected that approximately half of this difference is accounted for in the reduction of the assembly burnup due to uncertainty in the reactor records. However, if the assembly was at the low end of the reactor record uncertainty, the maximum error in burnup would be 10%. Since about 30% of the change in reactivity due to burnup is from fission products, this unexpected event is well within the available safety margin.

As stated in the first paragraph of this section, the measurement rejection requirement should be achievable with the current state of the art. Five percent is an engineering approximation of the uncertainty in both of the reactor records and measurement systems. Using this estimate, it would appear that deviations of greater than 10% between the measurement and reactor records would be unlikely and a basis for rejection.

6.4 MEASUREMENT SYSTEM DESIGN REQUIREMENTS

This section describes design requirements for burnup measurement systems. Any measurement system which meets these design requirements can be used to verify the burnup of fuel assemblies prior to loading in a burnup credit package. Burnup measurement systems fall into two broad classes, herein termed "dependent," and "independent." Dependent systems (e.g., gross neutron detection systems) rely on knowledge of the reactor record burnup values for a set of assemblies for a calibration. Therefore, these systems cannot truly "measure" burnup independently. The primary use for such systems is detection of "outlier" assemblies which for some reason have a radiation signature at odds with their reactor record burnup value. Independent measurement systems (e.g., gamma spectrum detection systems) are capable of performing a true independent measurement of assembly burnup, without reliance on reactor records, using the gamma emission signatures fission products (principally cesium isotopes).

These design requirements are performance-based; the operating principle and design details for the measurement system are not prescribed, and thus are left to the measurement system designer. However, because of the fundamental differences between dependent and independent measurement systems, specific requirements are developed for each system type. Where appropriate, a justification is provided after the requirement, which describes the rationale for the requirement. The requirements are detailed in the following sections.

6.4.1 Accuracy Requirement

6.4.1.1 Dependent Measurement Systems

Dependent measurement systems measure a neutron and/or gamma count rate, and plot count rate as a function of reactor record burnup for each assembly to generate a calibration curve. For dependent systems, a calibration curve of the following form is used to correlate the neutron counts to the reactor record burnup:

$$y_{counts} = a + bx_{reac} \text{ where} \qquad \text{Eq. 6-1}$$

a and b are constants, y_{counts} is the count rate (or, for neutron detection systems, typically the logarithm of the neutron count rate), and x_{reac} is the reactor record burnup value (or, for neutron detection systems, typically the logarithm of the reactor record value). Constants a and b are determined using standard linear regression techniques, following measurement of a group of assemblies.

The burnup uncertainty of dependent measurement systems is most conveniently stated in terms of a count rate predication band. (Note that for dependent measurement systems, the count rate prediction band incorporates both reactor records errors and intrinsic measurement system errors.) For dependent measurement systems, the count rate for a particular assembly should not differ from the calibration line by more than the following amount:

$$\text{Prediction Band Width (count rate)} = t_{0.025,n-2}\{[(n+1)/n+(x_i - x_{avg})^2/S_{xx}]SS_R/(n-2)\}^{0.5} \qquad \text{Eq. 6-2}$$

where,

$t_{0.025,n-2}$ is the t-distribution statistic bounding 95% of distribution for n degrees of freedom (two-sided distribution),
n is the number of assemblies in a calibration run,
x_i is the x_{reac} (burnup or log of burnup) for assembly I,
x_{avg} is the average of the x_{reac}'s for all assemblies in a calibration run,
$S_{xx} = \sum(x_i - x_{avg})^2$,
$SS_R = \sum(y_i - y_{fit})^2$,
y_i is the count rate (or log of the neutron count rate) measured for assembly I.
y_{fit} is the value from equation 6-1 for assembly I.

Since, for dependent measurement systems, prediction band width on uncertainty depends on the number of assemblies measured, an appropriate bound on the band width is required to ensure an adequate sample size for the calibration curve. Thus, dependent measurement systems must demonstrate, via analysis and confirmatory testing, that the following criterion can be met:

$$\text{Prediction Band Width (converted to burnup units) / Assembly Burnup} < 0.1$$

Where the Prediction Band Width is given in Equation 6-2.

The 10% requirement on the prediction band width is consistent with the 10% value used as a rejection criterion.

If reactor record values are such that they overestimate low burnup assemblies and underestimate high burnup assemblies, the neutron count rate vs. reactor record burnup calibration line could be unconservatively tilted, giving potentially erroneous results. To mitigate this effect, *either* of the following may be done:

a. Limit the set of assemblies used to develop a particular calibration line as follows: Ensure that the maximum and minimum reactor record burnup values for the set differ by at most

10 GWd/MTU. This minimizes the range of the calibration line, hence minimizing the effect of any unconservative tilt in the line.

or,

b. Establish via analysis the expected range for the calibration line's slope, and limit the slope to values within that range.

Ten GWd/MTU represents roughly one cycle of burnup. It is expected that assemblies within this burnup range have similar reactor power histories, and hence any reactor record calculational biases would be expected to be similar for such assemblies, and therefore would not cause a significant tilt in the calibration line. 10 GWd/MTU is also a large enough range to allow for an appropriately large calibration set. Establishing the expected slope range by analysis would appropriately bound the tilt which could be caused by reactor records biases.

6.4.1.2 Independent Measurement Systems

Independent measurement systems should demonstrate, via analysis and confirmatory testing, the uncertainty associated with a single assembly-average burnup measurement. That uncertainty should be 10% or less. This is consistent with the rejection criteria.

6.4.2 Correct Horizontal Average

The measurement system should account for the potential variation in burnup across the cross-section of the assembly, and the effect of such a variation on the measurement value should be quantified. Such an effect should not be large enough to cause the accuracy requirements to be exceeded. These types of radial effects shall be mitigated by measuring at least two opposing sides of the assembly, at the same assembly height (e.g., the assembly midplane).

6.4.3 Operational Considerations

The measurement system operating/calibration procedures shall detect and adjust for variations in the system and/or the environment which could affect the measurements, and/or the detector accuracy. Parameters which could affect measurements include, but are not limited to:

A. Detector electronic drift
B. Detector positioning
C. Pool boron concentration, temperature, and water purity
D. Counting time

6.4.4 Characteristics of Assemblies To Be Measured

Prior to measurement of assemblies with a particular set of characteristics, the measurement system shall be qualified, via analysis and/or qualification testing, to measure assemblies with

those characteristics to the accuracies specified in Section 6.4.1. Characteristics for which a measurement system should be qualified include:

A. Burnup Range
B. Initial Enrichment Range (Prior to measurement of assemblies with variable initial enrichments, the measurement system should be specifically qualified for such assemblies.)
C. Cooldown Range
D. Nominal Dimensions
E. Assembly Design Type

Note that a particular measurement system need not be qualified to measure the entire set of assemblies within the scope of this burnup credit methodology.

6.4.5 Analysis Tools

Analysis tools (e.g., computer codes) used to calculate assembly burnup values based on detector responses should be appropriately benchmarked, qualified, and up-to-date. Justification should be provided for why each analysis tool was used, including information on benchmarking and qualification which was performed for the tool. In particular, neutron detection systems should use an appropriately benchmarked and validated code to calculate Cm-244 production as a function of burnup. Neutron measurement systems should also account for neutron sources other than Cm-244, or should justify ignoring them.

6.4.6 Pool Compatibility

The process and equipment used for the verification measurement should be compatible with normal operations in spent fuel pools.

6.5 SUMMARY

Generic physical implementation and administrative control issues related to loading burnup credit packages are addressed by a generic burnup credit package loading process. The generic process provides the necessary control over nuclear criticality safety practices associated with loading burnup credit packages. The loading process relies on reactor records to establish fuel assembly loading qualification status and an independent burnup verification procedure to detect errors in the burnup records and ensure proper assembly selection. The proposed administrative controls and independent burnup verification procedure provide a high level of assurance that misloading of unqualified fuel assemblies will not occur.

This topical report is specifically seeking NRC acceptance of the use of reactor records (with reactor record burnup uncertainties accounted for by the utilities) to confirm fuel assembly

compliance with burnup credit analysis assumptions and parameters, and for the use of a measurement system which meets the requirements of Sections 6.3 and 6.4 to verify proper assembly selection prior to loading burnup credit packages. Reactor records, with burnup uncertainties accounted for, will be used in burnup credit SNF package loading procedures as the basis for assigning fuel assembly characteristics important to criticality safety. The specific fuel assembly parameters that must be assigned include: a) fuel assembly design type, b) initial enrichment, c) average burnup, d) cooling time following final reactor discharge, e) axial power shaping status, f) burnable absorber status, and g) intact configuration status. The measurement system is applied in the loading procedure to verify that the correct reactor records have been assigned to the proper assembly, as identified by a unique assembly identifier stamped on fuel assembly hardware.

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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. Reactor records for assembly enrichment and burnup are used to demonstrate that the loading criteria are satisfied. This assembly qualification process is augmented by a physical measurement that confirms proper assembly selection and reactor record assignment prior to declaring an assembly qualified for loading into a specific burnup credit package.

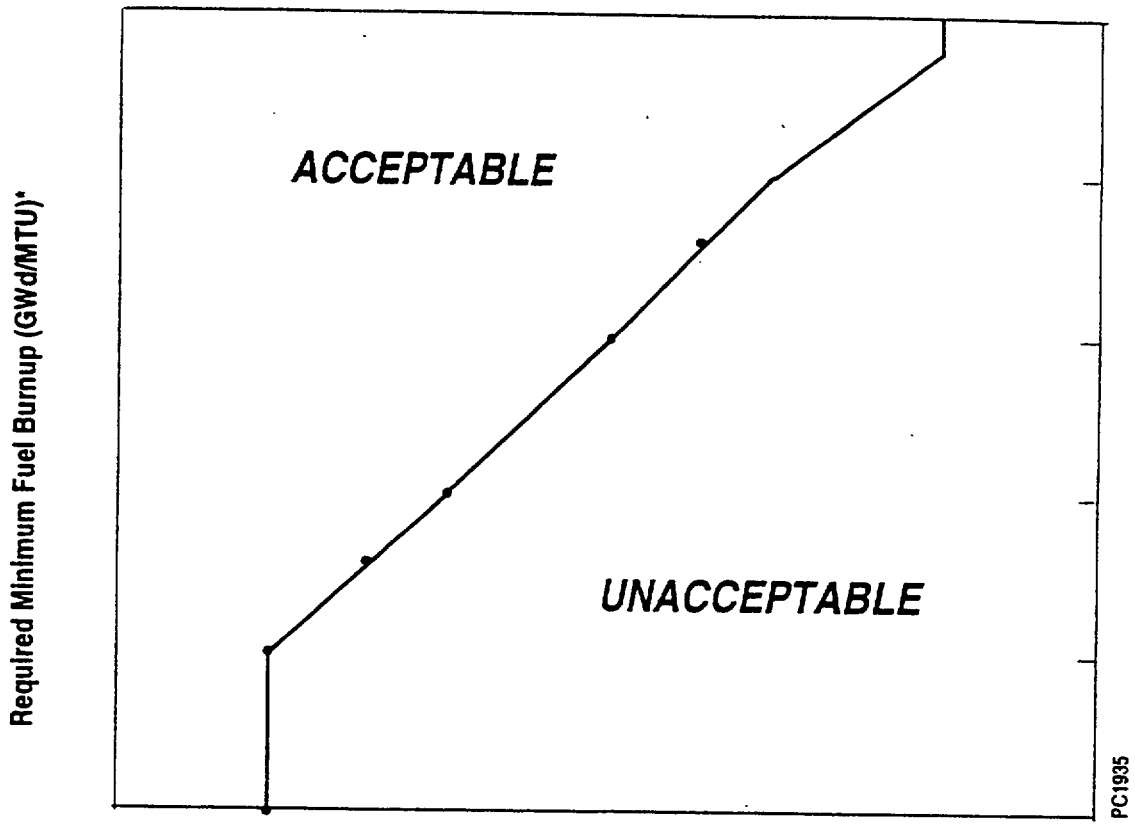
7.2 REVIEW OF THE BURNUP CREDIT STEPS

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



*Initial Fuel Enrichment (U-235 wt%)***

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-236, 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 validated for use in calculating the isotopic concentrations. The biases, uncertainties, and correction factors used are presented in Chapter 2. The validation of this system allows for quick use of an available code package, as well as an example of how to properly use the data and method presented.

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
	Perform trending analyses against a spectral parameter, fuel initial enrichment, fuel outside diameter, and soluble boron concentration for each subset. Calculate the bias and uncertainty utilizing all significant trends observed.	3.2
	Calculate the Upper Safety Limit (USL) for the UO_2 and MOX subsets and take the most limiting (validated USL for SCALE 4.2 in this report).	3.3
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 correct with 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 by measurement. Measurement and reactor record burnup must be within 10%. When comparing the selected assembly burnup to the package loading curve, the reactor record burnup must be decreased by the utility declared uncertainty in those records.	6.1

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

Fifty-seven critical experiments were selected to establish the bias over the anticipated range of PWR burnup credit package conditions. They span the range of applicability of the various parameters associated with a spent fuel shipping package. Table 3-2 in Chapter 3 provides the range of applicability matrix for the benchmark cases. The experiments consist of 21 UO_2 criticals (including two gadolinium criticals) and 36 mixed oxide configurations. The 19 UO_2 criticals without gadolinium were used to assess the computational bias associated with the various parameters affecting SNF package criticality (e.g., spacing, supplemental neutron absorbers, and reflector materials). Table 3-1 lists the parameters examined in these critical configurations for

the benchmarking process. Two UO_2-gadolinium experiments were evaluated to assess the bias associated with criticals with a harder neutron spectrum. The two experiments provide a well characterized experimental configuration with a known amount of gadolinium. The bias associated with fuel containing higher-order actinides may be obtained from MOX critical experiments. Thirty-six mixed-oxide criticals are also listed in Table 3-1. Chapter 3 provides references for the data required to analyze all the experiments listed in Table 3-1.

The criticality validation method developed combines biases, uncertainties, and an administrative safety margin to arrive at a USL for k_{eff}. This method of treating the criticality calculation bias has two primary components. A lower prediction band is defined by the 95% confidence level for a single future calculation. The prediction band width accounts for the statistical uncertainty in the bias. An administrative safety margin of 5% Δk_m is added to establish the USL, which becomes the bounding value for the criticality safety criterion. Although the use of a USL differs from the current practice of an explicit bias and criticality criterion, it provides a statistically sound method of establishing the bias as a function of any parameter while incorporating an additional safety margin that is consistent with the current practice.

With the experimental data established and the method developed, the CSAS criticality sequences of SCALE 4.2 using the 27BURNUPLIB cross section set were demonstrated to be valid for burnup credit SNF package criticality analyses. The USL as a function of the Average Lethargy for Absorption (ALA) is presented. This USL may be used for any package analysis using the validated code system (SCALE 4.2 with 27BURNUPLIB).

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. If the package has large margins, k_{eff} bias curves are provided that allow axially uniform analysis. 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 used in the analysis. Cooling times longer than the minimum specified are conservative for the first 100 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 includes a measurement technique to verify that the reactor records specified for a specific assembly correlate with the measured neutron or gamma emissions for the assembly. Chapter 6 describes the impact of burnup credit on the loading process. It discusses the enhanced procedures necessary to incorporate the verification measurement before package loading. The verification measurement reduces reliance on administrative controls and provides sufficient additional

protection against misloading to satisfy ANSI/ANS 8.1. A maximum disagreement between the measurement of burnup and burnup records is set at 10%. A set of minimum specifications for the measurement device is presented in Chapter 6.

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 all current commercial PWR fuel except that with integral fuel burnable absorbers. The critical experiments contained configurations with a wide variety of supplemental absorbers (integral and external to the fuel assemblies), reflectors, and pin spacings. 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 50 GWd/MTU. SNF with an assembly average burnup greater than 50 GWd/MTU shall be treated as having a burnup of 50 GWd/MTU for the purposes of this methodology.

   The highest burnup in the chemical assays was 46 GWd/MTU. There is sufficient data to allow burnups much greater than 50 GWd/MTU by extrapolation of trends, however, it is expected that burnup credit for burnups beyond 50 GWd/MTU will not be needed.

2. Enrichments above 5 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.

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 100 years.

   Cooling times less than 1 year are not of interest to current burnup credit concepts and therefore no effort was made to find the limits of applicability below 1 year. The 100 year limit is due to the reactivity increasing beginning sometime after 100 years.

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 requires 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 conservatism shown on Table 7-2 is large. Nevertheless, since strong documentation of individual fission products' worth is not available at this time, credit cannot be taken for fission products, and thus negative reactivity is present that is not taken credit for. Although fission product yields can be measured, the transmutation in the reactor has little experimental verification, and thus fission products' concentrations cannot be easily predicted.

Therefore, although obviously present in SNF providing considerable negative reactivity, fission products are not 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 Conservatism (% Δk)	Bounding Depletion Parameters Conservatism (% Δk)	Isotopic Correction Factors Conservatism (% Δk)	Added Conservatism (% Δk)
3.0	15	7.8	1.1	1.8	10.7
	30	12.2	3.1	2.4	17.7
	45	15.2	5.2	3.1	23.5
3.6	15	7.5	0.8	1.6	9.9
	30	11.9	2.3	2.2	16.4
	45	15.2	4.4	2.9	22.5
4.5	15	7.1	0.4	1.4	8.9
	30	11.4	1.4	1.9	14.7
	45	15.0	3.0	2.6	20.6

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 errors.

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 easily noted that credit is taken for only half of the reactivity change.

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	8.7	45%
	30	34.5	16.9	49%
	45	46.6	23.1	50%
3.6	15	18.2	8.3	46%
	30	32.8	16.4	50%
	45	45.6	23.1	51%
4.5	15	16.5	7.7	46%
	30	29.9	15.2	51%
	45	42.5	21.9	52%

Tables 7-2 and 7-3 only review the conservatisms in the isotopic calculations and exclusion of the fission products. 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. The magnitude of this conservatism can be estimated as similar to the k_{eff} bias curves (Figures 4-7 to 4-9) in Chapter 4. This results in a few more %Δk conservatism.

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 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 SUMMARY OF NRC APPROVALS SOUGHT

This topical report seeks NRC concurrence that: 1) the data presented are sufficient to validate actinide-only burnup credit, 2) the method presented to provide a basis for using burnup credit is valid, and 3) the computer analysis methods used to demonstrate this method are validated for burnup credit analysis as set forth in this topical report. This topical report is specifically seeking NRC acceptance of the following:

1. That 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. That 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. That the SAS2H sequence of the SCALE 4.2 code system using 27BURNUPLIB cross sections has been validated and appropriate isotopic correction factors have been determined.

4. That the 57 criticality experiments selected are sufficient for validating computer codes for actinide-only burnup credit analysis.

5. That trending analyses on the effect on k_{eff} due to variations in spectrum, initial enrichment, pellet outside diameter, and the soluble boron concentration are adequate.

6. That the method of determining the upper safety limit is adequate.

7. That the use of the developed USL with SCALE 4.2 code system with the 27BURNUPLIB and with a 0.05 Δk_m administrative safety margin is acceptable to perform actinide-only burnup credit criticality safety calculations in SNF package design.

8. That 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.

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

10. That the reactivity of the spent fuel is maximized by setting the fuel temperature to the maximum pellet averaged temperature.

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

12. That the method presented for determining optimum moderation in the SNF package is acceptable.

13. That the use of the selected limiting axial burnup profiles for burnup credit conservatively captures the end effects

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

15. That the method for developing the burnup loading criteria is adequate.

16. That the use of reactor records and the method of verifying proper assembly selection is acceptable.

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8. REFERENCES

Chapter 1

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1-3. T. L. Sanders, R. M. Westfall, and R. H. Jones, August 1987. *Feasibility and Incentives for the Consideration of Spent Fuel Operation Histories in the Criticality Analysis of Spent Fuel Shipping Casks*, Sandia National Laboratories, SAND87-0151.UC-71.

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APPENDIX A

EXAMPLES OF BURNUP CREDIT ANALYSIS

This Appendix demonstrates the application of the methodology described in Chapters 2 through 7. The demonstration is accomplished through contributions from cask vendors. All cask vendors with a license or license application for an SNF cask were invited to provide a sample calculation of their cask using actinide-only burnup credit. The methodology presented in the body of this report was still undergoing modification when it was provided to the cask vendors, so their sample calculations deviate in some degree from the method as now documented in this Topical Report. DOE supplied to the vendors isotopic data, a k_{eff} bias curve, and values for the horizontal burnup gradient. All of these have subsequently been modified. One vendor developed his own isotopic correction factors based on only a subset of chemical assays. This allows a good approximation of future results but should be viewed as only a demonstration since all the assays would be required for a license submittal. Because there have been small changes since the vendors began their work, it is important to not to use any of the data from the appendix when doing burnup credit analysis. However, upon review, the changes result in only slightly less burnup credit than that showed by the contributed sample calculations. The loading curves are close approximations and provide a good estimate of the impact of burnup credit on these products.

The cask vendors provided the following appendices at their own expense and DOE is grateful for their participation. Appendix A.1 is on the GA-4 cask. Appendix A.2 is on the Holtec HiStar-32. Appendix A.3 is on the TN-40. Finally, Appendix A.4 is on the Vectra MP-187. The appendices are in alphabetical order. The appendices are printed here exactly as provided by the vendors.

A.1 GA-4 LEGAL WEIGHT TRUCK CASK

A.1.1 Cask And Criticality Model Description

General Atomics (GA) has submitted a licensing application for the GA-4 cask that is currently being reviewed by the NRC Spent Fuel Project Office. The application has a fresh fuel assumption for the criticality analysis. The maximum enrichment permitted for shipping in the cask is 3.1 wt % with a maximum of 35 GWd/MTU burnup and 10-year cooling time or 45 GWd/MTU burnup and 15-year cooling time. Carrying the higher enrichment fuels at full capacity requires burnup credit for the criticality analysis.

Figure A.1-1 shows the GA-4 legal weight truck cask that has a capacity of up to four PWR fuel assemblies. GA designed the cask to maximize the authorized contents by means of a non-circular cask cross section. The shaped containment boundary and depleted uranium gamma shield fit closely around the array of spent fuel assemblies. A cylindrical shell surrounding the cask contains a neutron shield. A fixed stainless steel fuel support structure (FSS) separates the fuel assemblies and contains solid pellets of enriched boron carbide (B_4C) in radially drilled holes for criticality control. The B_4C pellets have a B-10 loading of 1.62 g/cm^3, allowing a compact array of fuel elements. The design uses two diameters of B_4C pellets. There are 141 in. of large diameter B_4C pellets in the middle region of the FSS and 9.5 to 10 in. of smaller diameter B_4C pellets at each end. The impact limiters utilize weight efficient aluminum honeycomb. The cask external dimensions are 39.75 in. diameter, 187.76 in. long without impact limiters and 233.95 in. long with impact limiters. The four 8.78-in. square by 167.26-in. long fuel cells provide sufficient space for all but the extra long PWR fuel assemblies. The Safety Analysis Report for Packaging (SARP), GA Document 910469, Revision D, submitted to the NRC has a more detailed description of the cask along with detailed drawings.

GA models the cask with a square cross section because geometry limitations in KENO V.a of the CSAS25 module in SCALE-4.3 prevent modeling the exact cross section. The slight increase of DU in the corners introduces additional fission reactions in the DU and also reflects more neutrons back into the system, making this assumption conservative.

The cask analytical model represents a full-height and 1/4-radial cross section of the cask. Figure A.1-2 illustrates the cask model used for criticality analysis. For this sample problem, GA uses the W 17x17 Std fuel assembly. We model the W 17x17 Std fuel assembly as a 17x17 array comprising (1) 264 fuel rods, including fuel, gap and cladding, and (2) 25 water holes. Table A.1-1 shows the fuel assembly model parameters. Figure A.1-3 is a cross sectional map of the fuel assembly as modeled. We minimize the assembly-to-assembly pitch (i.e., the assemblies are pushed to the center of the cask), to represent the most reactive configuration in the cask. To be conservative, we include the water holes as compared to modeling the entire 17x17 array filled with fuel. We model the B_4C with minimum pellet stack length and diameter in the center of the maximum diameter holes.

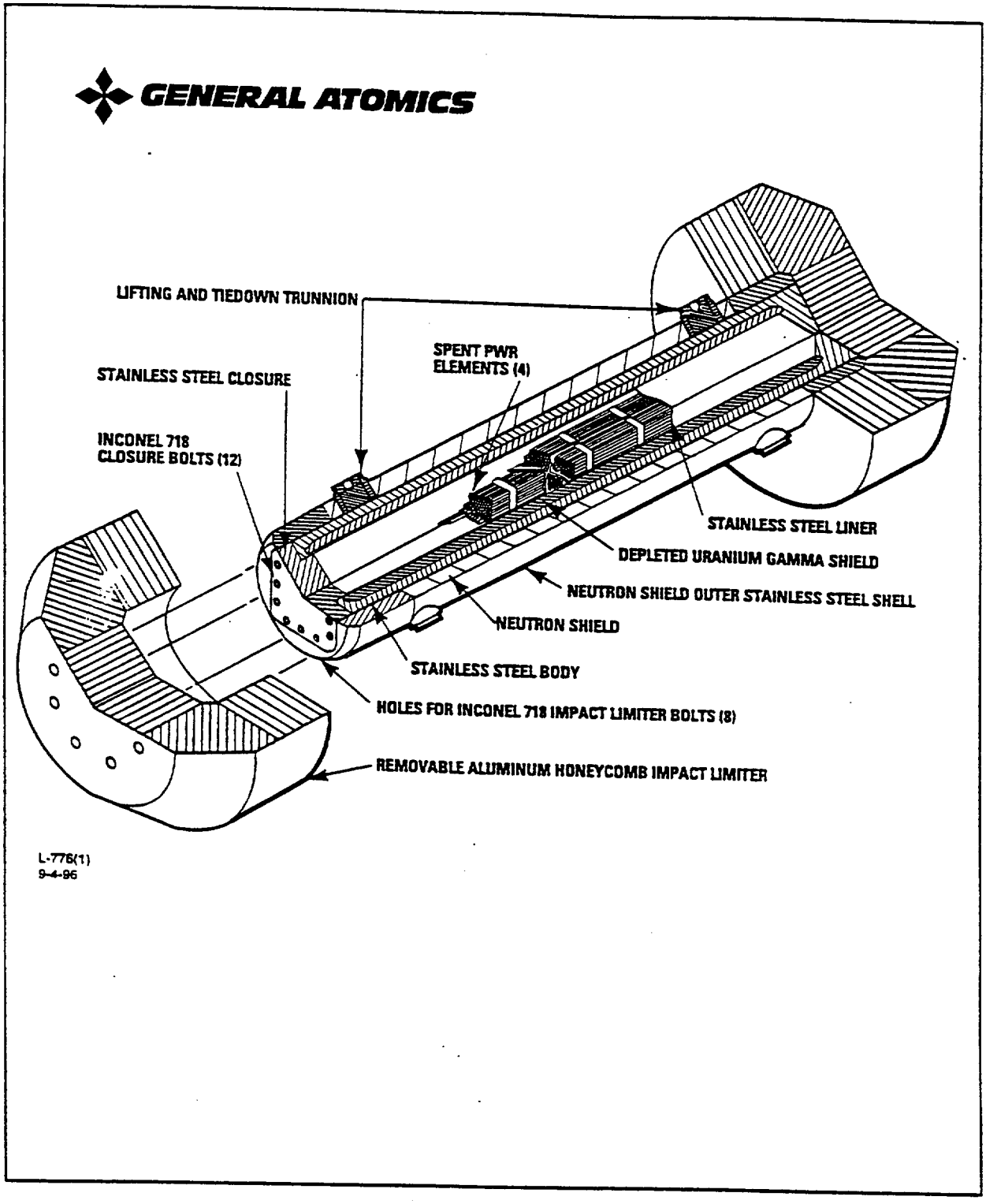


Figure A.1-1 GA-4 Legal Weight Truck Cask

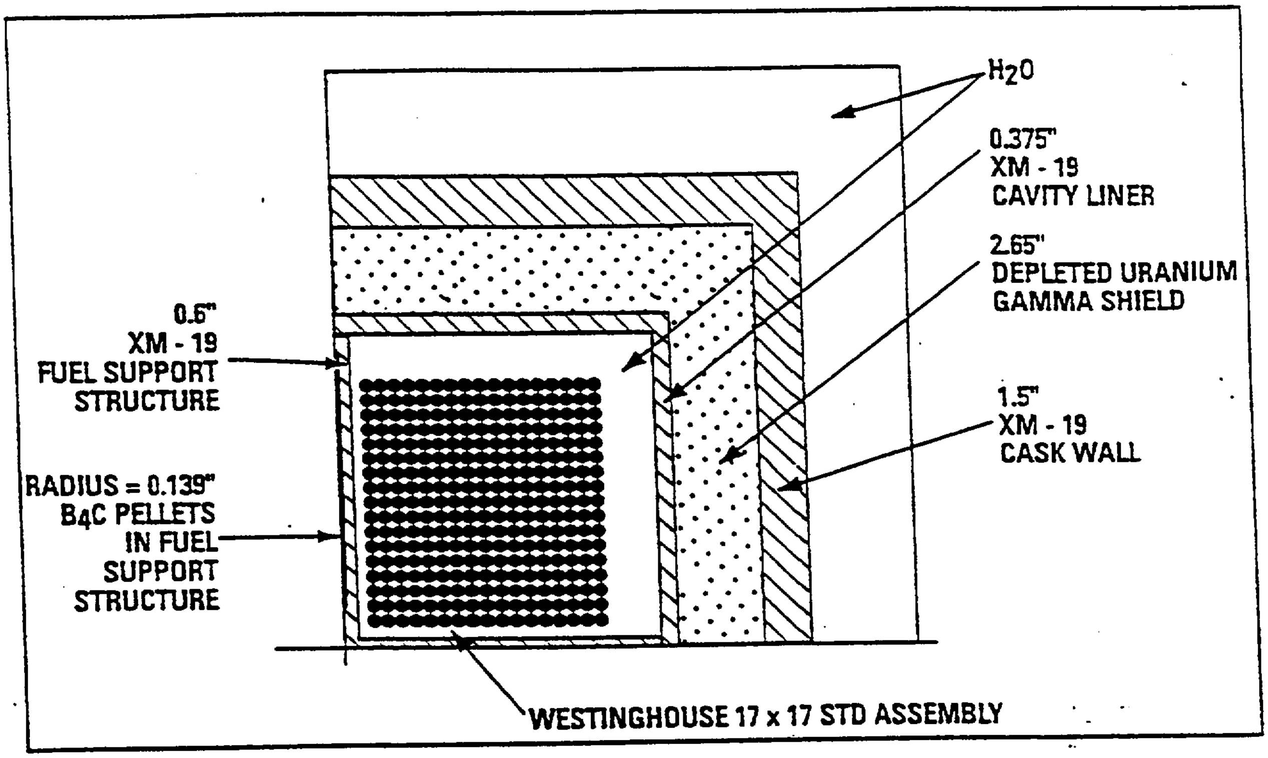


Figure A.1-2 Cross Section View of the KENO calculational model

```
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 1
1 1 1 2 1 1 1 1 1 1 1 1 1 2 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 2 1 1 1 1 1 1 1 1 1 2 1 1 1
1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1

1 = Fuel Rod
2 = Water Hole
```

Figure A.1-3 Cross Section View of the W 17 x 17 KENO calculational model

The Oak Ridge National Laboratory (ORNL) SCALE code package contains a standard material data library for common elements, compounds, and mixtures. This data library contains the materials used for the cask analysis. Water represents the neutron shield in the analytical model. The model does not include the neutron shield outer shell. The DOE computerized database contains material data for the fuel assemblies.

Table A.1-1 W 17x17 Fuel Assembly Model Parameters

Description	Value
Number of fuel rods	264
Number of water holes	25
Pitch, in.	0.496
Cladding o.d., in.	0.374
Cladding thickness, in.	0.0225
Pellet o.d., in.	0.3225
Active height, in.	144
Assembly width, in.	8.432
Pitch to rod diameter ratio	1.33
UO2 density, % TD	95
Linear U density of assembly, kg/cm	1.27
Moderator volume fraction	0.554
Cladding material	Zircaloy

Table A.1-2 provides a complete list of all the relevant materials used for the criticality evaluation. The material densities for the B4C represents 90% of the minimum B4C (96% enriched, 96% theoretical density pellets) poison content in the basket.

Table A.1-2 Model Material Data

Material	Density g/cm^3 ($lb/in.^3$)	Element	Physical wt %	Modeled Atom Density (atoms/b-cm)
B_4C	2.19 (0.0791)	B-10	73.91	9.760E–2
		B-11	3.08	3.698E–2
		C	23.01	2.532E–2
XM-19 (SS-304)	7.92 (0.286)	Cr	19.0	1.743E–2
		Mn	2.0	1.736E–3
		Fe	69.5	5.936E–2
		Ni	9.5	7.721E–3
Water	1.0 (0.0361)	H	11.1	6.677E–2
		O	88.9	3.338E–2
Depleted uranium	19.05 (0.688)	U-235	0.3	1.464E–4
		U-238	99.7	4.805E–2
Fuel W 17x17 UO_2 4.5% enriched	10.41 (0.3760)	U-234	0.035	9.84828E-6
		U-235	3.966	1.04557E-3
		U-236	0.019	4.64700E-6
		U-238	84.122	2.21749E-2
		O	11.858	4.64700E-2
Zircaloy	6.44 (0.2326)	Zr	100.0	4.25156E–2

A.1.2 Isotopic Validation

Since GA uses the SCALE program and the 27BURNUPLIB, Chapter 2 of this document provides the necessary isotopic validation. Chapter 2 contains the methodology that was used to calculate the isotopics and the correction factors that were used in this sample calculation.

A.1.3 Criticality Validation

Since GA uses the SCALE program and the 27BURNUPLIB, Chapter 3 of this document provides the necessary criticality validation. Chapter 3 contains the methodology used to develop the upper safety limit (USL). USL is defined as

for $ALA \leq 17.58$,
$USL = .8663 + 4.45 \times 10^{-3}(ALA) - 1.665 \times 10^{-3}\sqrt{23.402 + (ALA - 18.54)^2}$,

and

for $ALA \geq 17.58$,
$USL = 0.9381$,

where ALA = Average Lethargy for Absorption (calculated using program listed in Table A.1-3).

A.1.4 Limiting Parameters

Chapter 4 discusses many of the limiting parameters required for input to the SAS2H isotopic concentration generation process. The analysis to generate the isotopic concentrations uses the bounding parameter approach found in Chapter 4.

The physical data for the W 17x17 std fuel assembly are available in Report DOE/RW-0184. Table A.1-4 contains the parameters used in the criticality analysis.

Table A.1-3 Listing of Program to Calculate the Average Lethargy for Absorption (ALA)

```
      program ALAF
      dimension af(27),energy(28),xleth(27)
      character*30 filex
      character*14 line

C**ENERGY GROUPS
      energy(1)=2.00E7
      energy(2)=6.43E6
      energy(3)=3.00E6
      energy(4)=1.85E6
      energy(5)=1.40E6
      energy(6)=9.00E5
      energy(7)=4.00E5
      energy(8)=1.00E5
      energy(9)=1.70E4
      energy(10)=3.00E3
      energy(11)=5.50E2
      energy(12)=1.00E2
      energy(13)=3.00E1
      energy(14)=1.00E1
      energy(15)=3.05
      energy(16)=1.77
      energy(17)=1.30
      energy(18)=1.13
      energy(19)=1.00
      energy(20)=8.00E-1
      energy(21)=4.00E-1
      energy(22)=3.25E-1
      energy(23)=2.25E-1
      energy(24)=1.00E-1
      energy(25)=5.00E-2
      energy(26)=3.00E-2
      energy(27)=1.00E-2
      energy(28)=1.00E-5

C**ENTER FILE
   10 WRITE(*,*)'ENTER FILE NAME'
      READ(*,*) filex
      OPEN(UNIT=1,FILE=filex,status='old')

C**COMPUTE AVG LETHARGY FOR EACH ENERGY GROUP
      DO 20 j=1,27
        xleth(j)=(energy(j)*(alog(2E7/energy(j))+1)-energy(j+1)*
     +    (alog(2E7/energy(j+1))+1))/(energy(j)-energy(j+1))
   20 CONTINUE

C**READ ABSORPTION RATES FROM SCALE OUTPUT FILE
      DO 110 l=1,100000
        read(1,1001) line
        if (line.eq.'GROUP  FISSION') GOTO 120
  110 CONTINUE
  120 read(1,*)
      DO 130 j=1,27
        READ(1,*)
        read(1,*)x1,x2,x3,x4,af(j)
  130 CONTINUE
      CLOSE(UNIT=1)
```

Table A.1-3 Listing of Program to Calculate the Average Lethargy for Absorption (ALA) (Continued)

```
c**CONVERT RATES TO FRACTIONS
      SUM=0.
      DO 210 j=1,27
        sum=sum+af(j)
 210  CONTINUE


      DO 220 j=1,27
        af(j)=af(j)/sum
 220  CONTINUE

c**COMPUTE ALA
      ala=0.
      DO 310 j=1,27
         ala=ala+xleth(j)*af(j)
 310  CONTINUE

c**OUTPUT
      WRITE(*,*)'file:  ',filex
      WRITE(*,*)'ALA: ',ala
      WRITE(*,*)
      WRITE(*,*)
      WRITE(*,*)'Another file? (1=yes)'
      READ(*,*)nflag
      IF(NFLAG.EQ.1) GOTO 10

 1001 FORMAT(a14)
      STOP
      END
```

Table A.1-4. Key Parameters for Criticality Analysis

Description	Parameter
Fuel Support Structure (FSS) Poison Material	B_4C
Minimum B_4C Rod Diameter (in.) Small Pellets (at top and bottom of FSS) Large Pellets (small pellets assumed in model giving a uniform axial poison distribution to be consistent with the end effects bias calculation in Ch. 4)	 0.278 0.426
B_4C Rod Pitch (in.)	0.5
Maximum Fuel Cavity Width (in.)	8.796
Fuel Type	W 17x17
Fuel Assembly Pitch	Minimum
Number of Fuel Rods Number of Water Holes	264 25
Fuel rod pitch (in.)	0.496
Fuel o.d. (in.)	0.329
Cladding Thickness (in.)	0.0225
UO_2 smear density (% TD)	95.0
Fuel Enrichment/Burnup (wt% U-235/GWd/MTU)	3.1/7.5 3.1/9.5 3.1/10 3.1/15 3.5/10 3.5/15 3.5/16 3.5/20 4.0/20 4.0/25 4.0/30 4.5/30 4.5/33 4.5/35
Cooling Time (Years)	5

In addition to the key parameters shown in Table A.1-4, the following conservative assumptions are also incorporated into the criticality calculations:

1. Omission of grid plates, spacers, and hardware in the fuel assembly.
2. No credit taken for residual burnable poisons in the fuel.
3. Water density at 1.0 g/cm^3.
4. Temperature at 20°C (293°K).
5. B_4C density assumed to be 86.4% of theoretical density which accounts for 4% manufacturing uncertainty and 10% margin.
6. No boron modeled in the neutron shield.
7. Only the actinide isotopes are used in the analysis.
8. Uniform axial poison used in model. The small pellets of B_4C were assumed for the entire height of the FSS instead of a combination of small and larger diameter pellets as described in Section A.1.1. This conservatism was added to be consistent with the calculation of the end effects bias shown in Chapter 4, which was calculated based on a uniform axial poison.
9. Uniform axial burnup profile used in analysis.

A.1.5 Construction Of Loading Curve

A.1.5.1 Fresh Fuel Intercept

Fresh fuel calculations are a series of CSAS25 (KENO) calculations using the cask model to determine the maximum fresh fuel enrichment that can be safely loaded into the cask. Issue D of the GA-4 Safety Analysis Report for Packaging, submitted to the NRC for review, contains this calculation for all 14x14 and 15x15 PWR fuel assemblies and shows that the intercept is at 3.1 wt % enrichment which means that all of these assemblies with an initial enrichment $\leq$ 3.1 wt% can be shipped without burnup credit. In Issue N/C of the GA-4 SARP, we showed that the most reactive assembly is the W 15x15 OFA and that it is more reactive than the W 17x17 Std assembly. Therefore, a fresh fuel intercept calculation was not repeated for this sample calculation of the W 17x17 Std assembly. Instead, we use the conservative value of 3.1 wt % determined for the most reactive assembly.

A.1.5.2 Spent Fuel Composition

GA performed a linear interpolation of the isotopic concentrations provided by DOE to determine the isotopic input to the CSAS25 calculation described in Section A.1.5.3. As stated in Chapter 4, a flat one-zone profile with the addition of the end effects bias is a conservative model.

A.1.5.3 Spent Fuel Criticality Calculations

The adjusted isotopics described in Section A.1.2 for a specific initial enrichment and burnup are used as input to CSAS25 to determine the reactivity of the total fuel cask model. To account for radial burnup variation (tilt) in each assembly, each assembly in the model has two burnup levels. For the four element array in the GA-4 Cask, it is obvious that the most reactive configuration is when the lowest burnup for each element is towards the center of the array. DOE has calculated the appropriate amount of radial tilt and is shown in Figure A.1-4. The tilt is 33% for burnup < 20 GWd/MTU, 20% for burnup ≥ 20 GWd/MTU and <40 GWd/MTU and 15% for burnup ≥ 40 GWd/MTU.

Using the CSAS25 results, the Average Lethargy of Absorption (ALA) is calculated using the computer code shown in Table A.1-3. The USL was determined using the formulas described in Section A.1.3. Table A.1-5 shows the input for one such calculation at 4.0 wt% initial enrichment and 25 GWd/MTU burnup. Table A.1-6 shows the results for the calculated enrichment and burnup pairs that went into the loading curve determination.

Figure A.1-4 plots the calculated k_{eff} + 1.645σ results from the CSAS25 calculations plus the end effects bias from Chapter 4, and the evaluated USL for the example case. The intercept of the USL and k_{eff} + 1.645σ + end effects bias determines the minimum burnup required to safely load the given initial fuel enrichment. Table A.1-7 shows the final required burnup along with the confirmatory calculated k_{eff} values at or below the intercepts. The confirmatory calculated k_{eff} for these results is close to and below the USL. Figure A.1-5 plots the final required burnup versus initial enrichment for W 17x17 assemblies. As explained in Chapter 5, only spent fuel assemblies with minimum burnup and initial enrichment above the loading curve are qualified for loading into the GA-4 cask. Assemblies that fall below the curve cannot be shipped in this cask. For information, we also show the distribution of all PWR fuel assemblies that fall within each burnup and enrichment range (1993 data, Ref. EIA Service Report, February 1995).

A.1.5.4 Low-Density Moderation Effects

As described in Chapter 4, criticality 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 Part 71.55). The procedure calls for evaluating the reactivity at reduced moderator density. For this sample calculation, this analysis is not performed since a water density of 1.0 is known to be limiting.

Table A.1-5 Input File for CSAS25 Calculation of 4.0 wt% Initial Enrichment and 25 GWd/MTU Burnup

```
=CSAS25
4 PWR ASSEM. W(17X17) STD, E=4.0, BU=25,
27BURNUPLIB  LATTICECELL
U-234    1 0 4.652-6 END
U-235    1 0 5.508-4 END
U-236    1 0 7.934-5 END
U-238    1 0 2.170-2 END
PU-238   1 0 7.712-7 END
PU-239   1 0 1.374-4 END
PU-240   1 0 2.744-5 END
PU-241   1 0 1.394-5 END
PU-242   1 0 2.315-6 END
AM-241   1 0 3.281-6 END
O        1 0 4.622-2 END
ZIRCALLOY  2 1.0 END
H2O  3 1.0 END
B4C  4 0.864 293.0 5010 96.0 5011 4.0 END
SS304  5 1.0 END
URANIUM  6 1.0 293.0 92235 0.3 92238 99.7 END
U-234    7 0 4.064-6 END
U-235    7 0 3.961-4 END
U-236    7 0 1.002-4 END
U-238    7 0 2.153-2 END
PU-238   7 0 2.127-6 END
PU-239   7 0 1.541-4 END
PU-240   7 0 4.336-5 END
PU-241   7 0 2.448-5 END
PU-242   7 0 6.749-6 END
AM-241   7 0 5.875-6 END
O        7 0 4.622-2 END
H2O  8 1.0 END
H2O  9 1.0 END
END COMP
SQUAREPITCH 1.25984 0.81915 1 3 0.94996 2 0.83566 8 END
MORE DATA
RES=7 CYLINDER 0.409575 DAN(7)=0.261107
END MORE DATA
FLAT FULL HEIGHT GA-4, 2/97, 90% FOR NRC WITH B4C
READ PARAM  TME=200.0 GEN=200 NPG=1000 FLX=NO FDN=NO
NUB=YES END PARAM
READ ARRAY
ARA=1 NUX=19 NUY=19 NUZ=1
FILL
      4 5 7 14R9 11 13
      6 17R1 15
      8 16R1 3 15
     10 5R1 2 2R1 2 2R1 2 3R1 2R3 15
     10 3R1 2 9R1 2 3R3 15
     10 12R1 5R3 15
     10 2R1 2 2R1 2 2R1 2 2R1 2 2R3 2 2R3 15
     10 10R1 7R3 15
     10  9R1 8R3 15
     10 2R1 2 2R1 2 2R1 2 2R3 2 2R3 2 2R3 15
     10  7R1 10R3 15
     10  6R1 11R3 15
     10 2R1 2 2R1 2 2R3 2 2R3 2 2R3 2 2R3 15
     10  4R1 13R3 15
     10 3R1 2 9R3 2 3R3 15
     10 3R1 2R3 2 2R3 2 2R3 2 5R3 15
     10 2R1 15R3 15
     12 1 16R3 15
     14 17R16 17
     END FILL
```

Table A.1-5 Input File for CSAS25 Calculation of 4.0 wt% Initial Enrichment and 25 GWd/MTU Burnup (Continued)

```
ARA=2 NUX=19 NUY=19 NUZ=1
FILL
      4 5 7 14R9 11 13
      6 17R18 15
      8 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     10 17R18 15
     12 17R18 15
     14 17R16 17
     END FILL
ARA=3 NUX=19 NUY=19 NUZ=1
FILL
      4 17R19 13
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     20 17R18 15
     14 17R16 17
     END FILL
ARA=4 NUX=1 NUY=1 NUZ=333
FILL
     5R23
     288R21
     34R22
     6R23
     END FILL
END ARRAY
READ BNDS -XY=MIRROR   END BNDS
READ GEOM
UNIT 1
CYLINDER 1 1 0.409575 2P0.635
CYLINDER 8 1 0.41783 2P0.635
CYLINDER 2 1 0.47498 2P0.635
CUBOID   3 1 4P0.62992 2P0.635
UNIT 2
CYLINDER 3 1 0.56900 2P0.635
CYLINDER 2 1 0.61470 2P0.635
CUBOID   3 1 4P0.62992 2P0.635
```

Table A.1-5 Input File for CSAS25 Calculation of 4.0 wt% Initial Enrichment and 25 GWd/MTU Burnup (Continued)

```
UNIT 3
CYLINDER 7 1 0.409575 2P0.635
CYLINDER 8 1 0.41783 2P0.635
CYLINDER 2 1 0.47498 2P0.635
CUBOID   3 1 4P0.62992 2P0.635
UNIT 4
CUBOID 5 1 4P0.381 2P0.635
UNIT 5
xHEMICYL+y 4 1 0.35306 1.25984 0.4572
XHEMICYL+Y 0 1 0.37084 1.25984 0.23622
CUBOID     5 1 1.25984 0.0 0.762 0.0 2P0.635
UNIT 6
YHEMICYL+X 4 1 0.35306 1.25984 0.4572
YHEMICYL+X 0 1 0.37084 1.25984 0.23622
CUBOID     5 1 0.762 0.0 1.25984 0.0 2P0.635
UNIT 7
XHEMICYL+Y 4 1 0.35306 1.25984 0.0
XHEMICYL+Y 0 1 0.37084 1.25984 0.0
CUBOID     5 1 1.25984 0.0 0.762 0.0 2P0.635
UNIT 8
YHEMICYL+X 4 1 0.35306 1.25984 0.0
YHEMICYL+X 0 1 0.37084 1.25984 0.0
CUBOID     5 1 0.762 0.0 1.25984 0.0 2P0.635
UNIT 9
XHEMICYL+Y 4 1 0.35306 2P0.62992
XHEMICYL+Y 0 1 0.37084 2P0.62992
CUBOID     5 1 2P0.62992 0.762 0.0 2P0.635
UNIT 10
YHEMICYL+X 4 1 0.35306 2P0.62992
YHEMICYL+X 0 1 0.37084 2P0.62992
CUBOID     5 1 0.762 0.0 2P0.62992 2P0.635
UNIT 11
XHEMICYL+Y 4 1 0.35306 0.7874 0.0
XHEMICYL+Y 0 1 0.37084 0.7874 0.0
CUBOID     5 1 1.25984 0.0 0.762 0.0 2P0.635
UNIT 12
YHEMICYL+X 4 1 0.35306 0.7874 0.0
YHEMICYL+X 0 1 0.37084 0.7874 0.0
CUBOID     5 1 0.762 0.0 1.25984 0.0 2P0.635
UNIT 13
CUBOID 5 1 2P0.46228 2P0.381 2P0.635
UNIT 14
CUBOID 5 1 2P0.381 2P0.46228 2P0.635
UNIT 15
CUBOID 3 1 2P0.46228 2P0.62992 2P0.635
UNIT 16
CUBOID 3 1 2P0.62992 2P0.46228 2P0.635
UNIT 17
CUBOID 3 1 4P0.46228 2P0.635
UNIT 18
CUBOID 3 1 4P0.62992 2P0.635
UNIT 19
CUBOID     5 1 2P0.62992 0.762 0.0 2P0.635
UNIT 20
CUBOID     5 1 0.762 0.0 2P0.62992 2P0.635
UNIT 21
ARRAY 1 0.0 0.0 -0.635
UNIT 22
ARRAY 2 0.0 0.0 -0.635
UNIT 23
ARRAY 3 0.0 0.0 -0.635
CORE 4 1 0.0 0.0 0.9652
CUBOID 3 1 23.2029 0.0 23.2029 0.0 424.815 0.0
```

Table A.1-5 Input File for CSAS25 Calculation of 4.0 wt% Initial Enrichment and 25 GWd/MTU Burnup (Continued)

```
CUBOID 5 1 24.1554 0.0 24.1554 0.0 424.815 0.0
CUBOID 6 1 30.8864 0.0 30.8864 0.0 424.815 0.0
CUBOID 5 1 34.6964 0.0 34.6964 0.0 452.755 -24.13
CUBOID 9 1 100.0 0.0 100.0 0.0 510.0 -75.0
END GEOM
END DATA
END
```

Table A.1-6 Spent Fuel Calculational Results - Actinides Only

Enrich-ment	Burnup (GWd/MTU)	K_{eff}	σ	End Effects Bias	ALA	K_{eff} + 1.645σ. + End Effects Bias	USL
3.1	7.5	0.9340	0.0014	0.0000	16.07	0.9363	0.9288
3.1	9.5	0.9263	0.0015	0.0000	16.08	0.9288	0.9288
3.1	10	0.9231	0.0015	0.0000	16.08	0.9256	0.9288
3.1	15	0.8966	0.0015	0.0000	16.06	0.8991	0.9287
3.5	10	0.9478	0.0014	0.0000	16.02	0.9501	0.9285
3.5	15	0.9248	0.0015	0.0000	16.02	0.9273	0.9285
3.5	16	0.9216	0.0014	0.0000	16.02	0.9239	0.9285
3.5	20	0.9061	0.0013	0.0000	16.01	0.9082	0.9285
4.0	20	0.9417	0.0015	0.0000	15.98	0.9442	0.9283
4.0	25	0.9200	0.0016	0.0028	15.97	0.9254	0.9282
4.0	30	0.8974	0.0015	0.0055	15.97	0.9054	0.9283
4.5	30	0.9274	0.0015	0.0055	15.93	0.9354	0.9280
4.5	33	0.9179	0.0014	0.0072	15.95	0.9274	0.9281
4.5	35	0.9086	0.0015	0.0083	15.94	0.9193	0.9281

Table A.1-7 GA-4 Cask Loading Curve Data for W 17x17 Fuel (Actinides Only)

Initial Enrichment (wt%)	Burnup (GWd/MTU)	Confirmatory Calculated k_{eff} + 1.645σ + End Effect Bias	ALA	USL
3.1	9.5	.9288	16.08	.9288
3.5	15	.9273	16.02	.9285
4.0	25	.9254	15.97	.9282
4.5	33	.9274	15.95	.9281

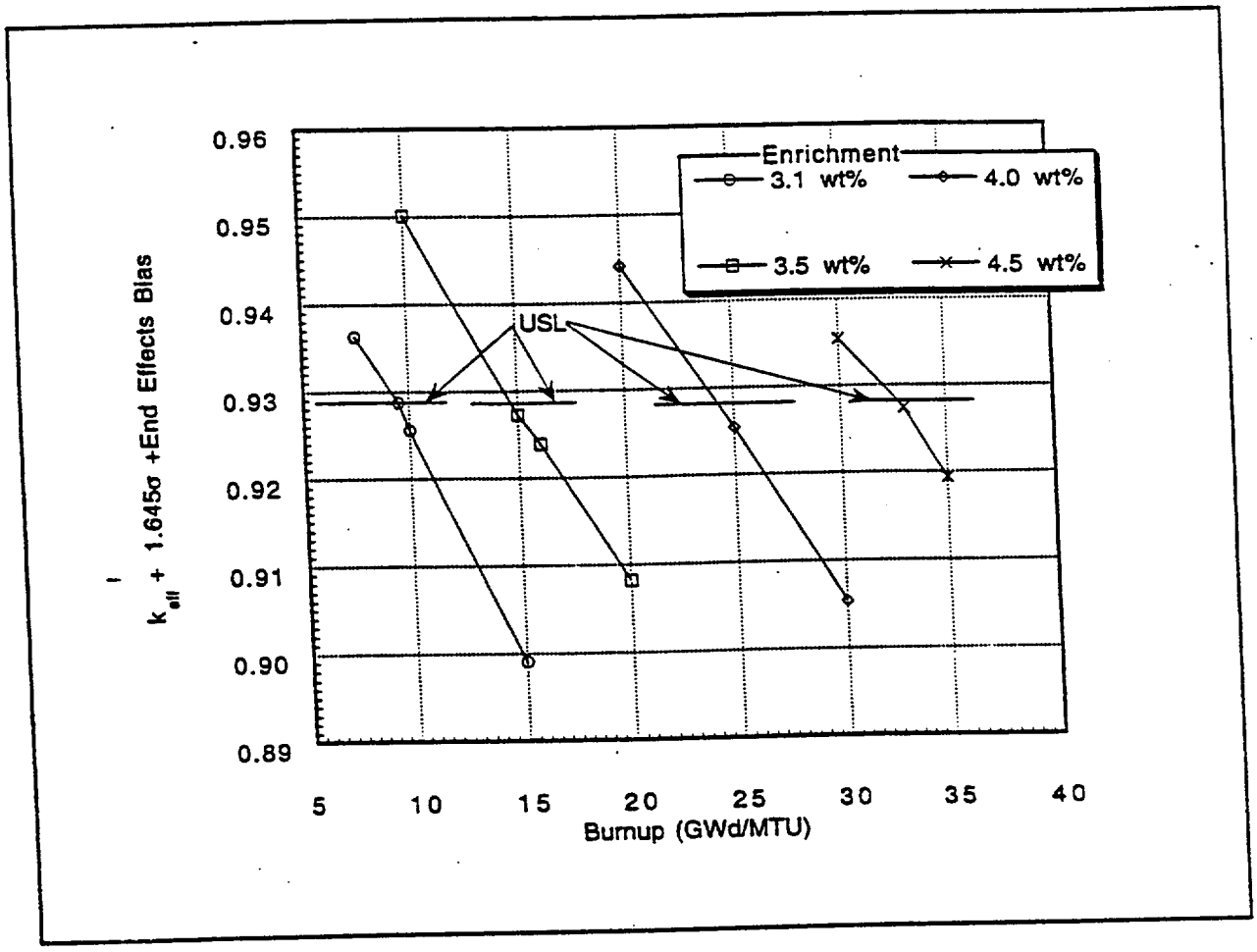


Figure A.1-4 Calculational Results Leading to the Generation of a Loading Curve

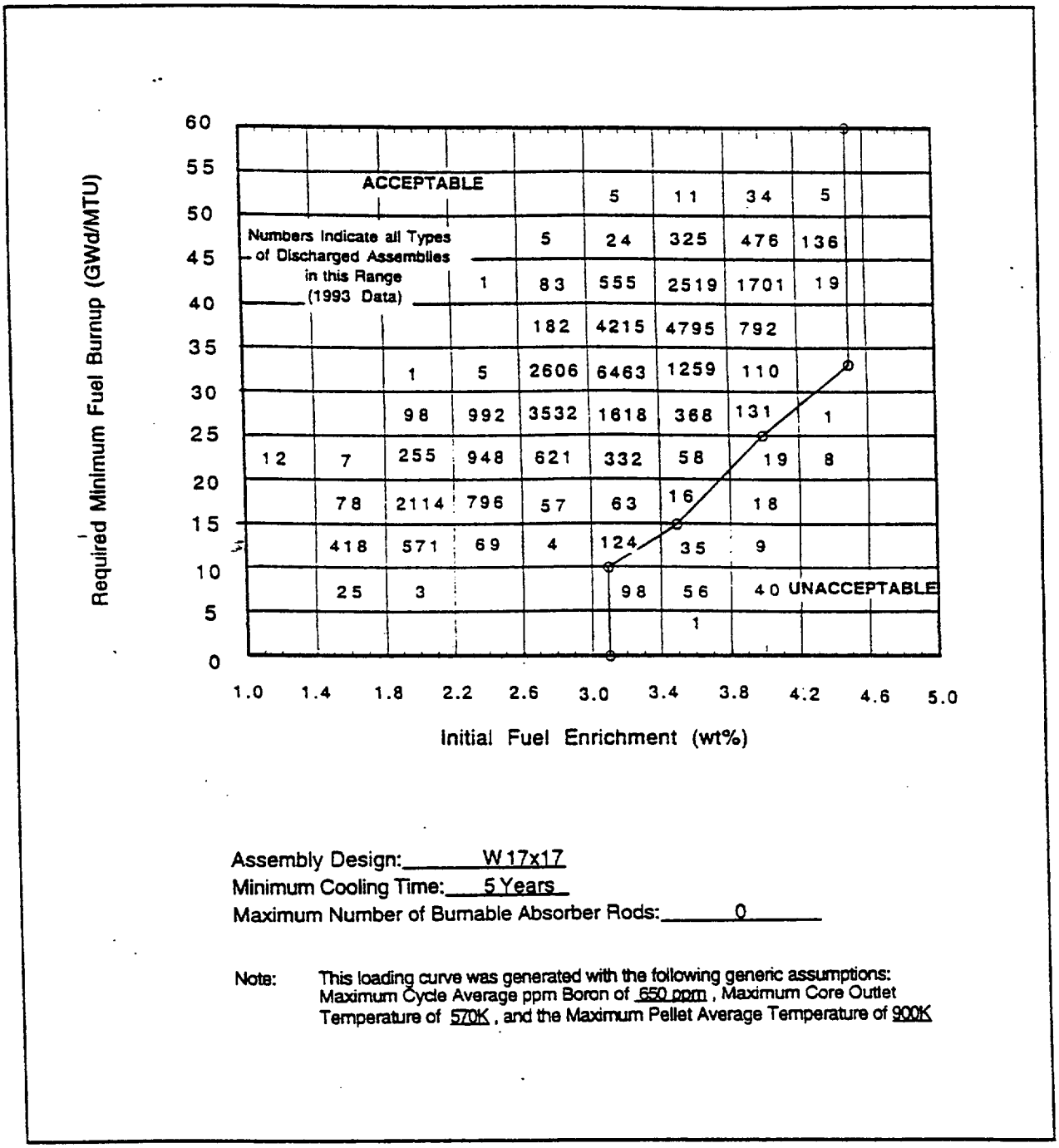


Figure A.1-5 Loading Curve for W 17x17 Fuel Assemblies in the GA-4 Cask

A.2 HOLTEC HI-STAR 100 / MPC-32

A complete demonstration of the actinide-only burnup credit methodology introduced in this topical report is performed using Holtec's HI-STAR 100 package with the MPC-32 canister. CASMO-3 is used for the isotopic depletion analyses, while MCNP is utilized for the criticality calculations. A sample loading curve is generated following the guidance in Chapters 2 thru 5. Simplifications in the analyses are introduced, yet the overall effect and importance of burnup credit are evident.

A.2.1 CASK MODEL DESCRIPTION

HI-STAR 100 (acronym for Holtec International Storage, Transport and Repository) is a spent nuclear fuel packaging designed to be in general compliance with the U. S. Department of Energy's design procurement specifications for multi-purpose canisters and large transportation casks. The annex "100" is a model number designation which denotes the system weighing in the range of 100 tons. The HI-STAR 100 System consists of a sealed metallic canister (MPC) contained within an overpack. It is designed to accommodate a wide variety of spent fuel assemblies in a single overpack by utilizing different MPCs. For this appendix, the MPC-32 is featured. This canister can contain a maximum of 32 PWR assemblies. Currently, the fresh fuel limit on enrichment for the MPC-32 is 1.9 wt% U-235. Figure A.2-1 depicts the HI-STAR 100 with two of its major constituents, the MPC and the overpack, in a cutaway view.

The MPC is a welded cylindrical structure with flat ends as shown in cross sectional view of Figure A.2-2. The MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring. The outer diameter and cylindrical height of the MPC are fixed to fit into the generic overpackage. The MPC provides the confinement boundary for the stored fuel.

The HI-STAR 100 is designed for both storage and transport. The HI-STAR 100 System's multi-purpose design reduces SNF handling operations and thereby enhances radiological protection. Once the SNF is loaded and the MPC and cask are sealed, the HI-STAR 100 System can be positioned on-site for temporary or long-term storage or transported directly off-site. The system's ability to both store and transport SNF eliminates repackaging.

The HI-STAR 100 System is a completely passive stand-alone storage system which provides SNF confinement, radiation shielding, structural integrity, criticality control, and heat removal independent of any other facility, structures or components. Further information on the HI-STAR 100 and the MPC-32 can be found in the Safety Analysis Report, Holtec Report HI-951251 (NRC Docket No. 71-9261).

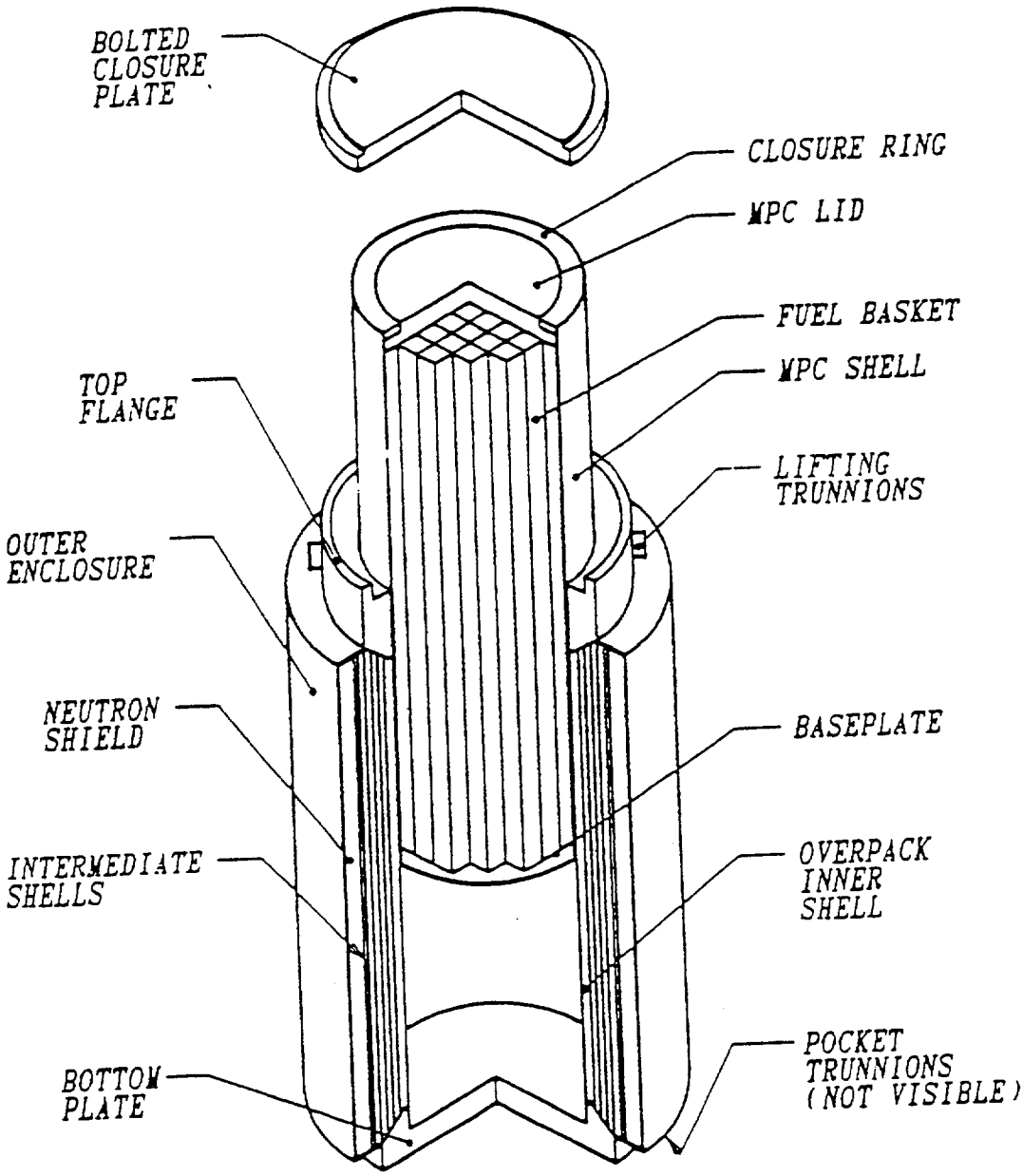


Figure A.2-1. HI-STAR 100 Overpack with MPC Partially Inserted

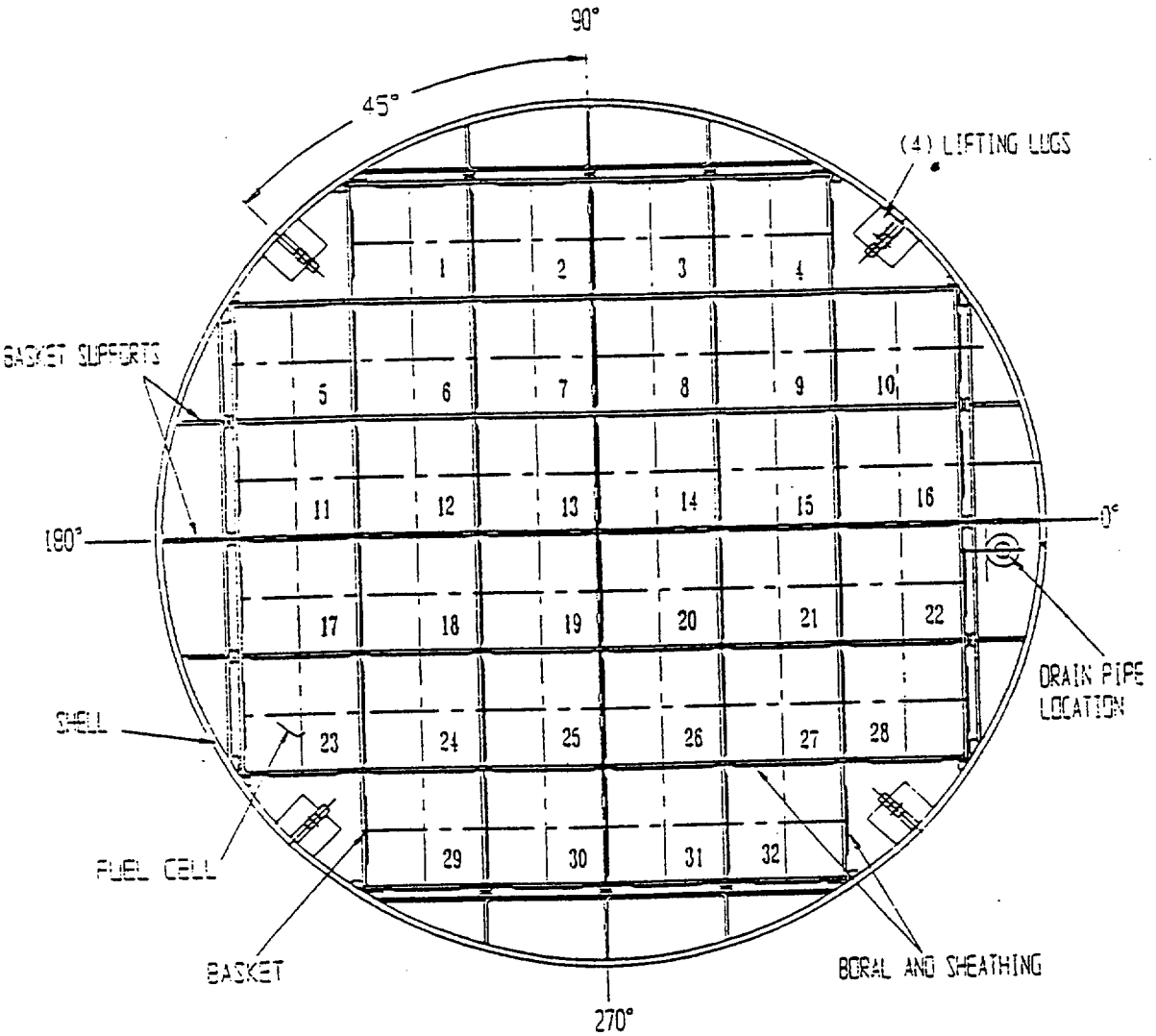


Figure A.2-2. MPC-32 Cross Section View

A.2.2 ISOTOPIC VALIDATION

CASMO-3, a two-dimensional transport theory code for fuel assemblies, is used for the depletion calculations. As required for the actinide-only burnup credit methodology, the code is validated by use of the benchmark set of chemical assays presented in Chapter 2. For the analyses presented in this appendix, a limited set of the assays are modeled; also, U-234 and Pu-238 are excluded. Based on the results for the limited set, the correction factors are generated. The computed correction factors are presented in Table A.2-1. For five of the isotopes, the measured to calculated ratios have statistically significant trends against burnup, and are treated accordingly.

Table A.2-1. Correction Factors for CASMO-3

Isotope	Correction Factor
U-235	max[1.0 + 0.00148*BU + 0.000707*SQRT(1.090E4 + BU^2), 1.0]
U-236	0.905
U-238	0.989
Pu-239	max[1.0 + 0.00180*BU + 0.000838*SQRT(1.090E4 + BU^2), 1.0]
Pu-240	0.944
Pu-241	max[1.0 + 0.00221*BU + 0.000751*SQRT(1.090E4 + BU^2), 1.0]
Pu-242	min[1.0 + 0.00604*BU - 0.00129*SQRT(9.90E3 + BU^2), 1.0]
Am-241	min[1.0 + 0.00221*BU - 0.000751*SQRT(1.090E4 + BU^2), 1.0]

A.2.3 CRITICALITY VALIDATION

Criticality analyses are performed with the MCNP code. Validation of the code is achieved by using forty-three of the criticals experiments in the benchmark set. Similar to the results presented in Chapter 3 for the SCALE 4.2 system, a trend is observed against the spectral parameter (the average lethargy for fission was used in these analyses) for the MOX subset. The resulting upper safety limit (USL) is,

$ALF \leq 17.42$: $USL = 0.8975 + 0.003052 * ALF - 0.001996*SQRT[14.81 + (ALF-17.84)^2]$

$ALF \geq 17.42$: $USL = 0.9429$

A.2.4 LIMITING PARAMETERS

Chapter 4 discusses the limiting parameters required for input to the isotopic depletion calculations. Values suggested by DOE are used for the analyses presented in this appendix, and are listed in Table A.2-2. For the criticality calculations, the calculational model used included multiple conservatisms due to material and fabrication tolerances and modeling limitations. End effects are included by adding the appropriate k_{eff} bias from the tables included in Chapter 4. Analyses on low-density moderation effects are not performed; a value of 1.0 has been previously shown to be limiting, and its used throughout the criticality calculations. For simplicity, the horizontal burnup gradient effects are neglected, which should be minimal for the MPC-32.

Table A.2-2. Limiting Parameters for Isotopic Calculations

Parameter	Value
Moderator Temperature	570 K
Fuel Temperature	900 K
Boron Concentration	650 ppm
Specific Power	60 MW/MTU

A.2.5 CONSTRUCTION OF LOADING CURVE

A sample actinide-only burnup credit loading curve is generated for the MPC-32. Analyses are performed for the Westinghouse standard 17X17 V5H assembly.

A.2.5.1 Fresh Fuel Intercept

The maximum fresh fuel enrichment is computed by executing the MPC-32 MCNP model at different enrichment values. To determine this limiting enrichment, the k_{eff} results from MCNP are augmented by 1.645σ and then compared to the fresh fuel (UO_2 subset) USL. The enrichment value at which a match is obtained is labeled the fresh fuel intercept. Table A.2-3 presents the results for these analyses. A value of 2.0 wt% U-235 is determined to be the fresh fuel limit on enrichment. The value is slightly larger than the 1.9 wt% U-235 fresh fuel limit for the MPC-32 due to different methodologies used to compute this limit.

Table A.2-3. Calculation Results for MPC-32 Maximum Fresh Fuel Enrichment

Enrichment (wt % U-235)	k_{eff}	σ	k_{eff} + 1.645 σ	USL
1.8	0.9075	0.0010	0.9091	0.9429
2.0	0.9387	0.0010	0.9403	0.9429

A.2.5.2 Spent Fuel Composition

Multiple CASMO-3 runs are executed at different burnup and enrichment values using the depletion limiting parameters. The calculated concentration values are adjusted by the correction factors and converted to the necessary units for input into MCNP.

A.2.5.3 Spent Fuel Criticality Calculations

Criticality calculations are performed to generate the burnup credit loading curve. At several enrichments, burnup values are selected to obtain a biased k_{eff} that matches the USL at the particular average lethargy for fission (ALF) for the cask model analyzed. The multiplication factors obtained from MCNP are augmented by 1.645 * σ and the corresponding k_{eff} bias due to the end effects. Sample MCNP calculational models are included in the HI-STAR 100 Safety Analysis Report (NRC Docket No. 71-9261).

Table A.2-4 and Figure A.2-3 show the results used for the generation of the loading curve, which is shown in Figure A.2-4. Analyses are performed on both five year cooled SNF and twelve year cooled fuel; both loading curves are included in Figure A.2-4. Since the analyses presented are only for demonstration purposes, multiple simplifications were used that resulted in a very conservative loading curve. A more detailed application of the burnup credit methodology on the MPC-32 would lower the loading curve as a result of performing axial analyses instead of assuming a generic end effect k_{eff} bias, and also due to longer cooling times as required for thermal considerations.

Table A.2-4. Criticality Calculations Results

Enrichment (wt % U-235)	Burnup (GWd/MTU)	k_{eff}	σ	k_{eff} bias	ALA	k_{eff} + 1.645σ + k_{eff} bias	USL
2.0	3	0.9559	0.0009	.0032	17.60	0.9605	0.9429
	8	0.9305	0.0009	.0084	17.50	0.9404	0.9429
2.5	15	0.9481	0.0011	.0158	17.39	0.9657	0.9428
	18.5	0.9265	0.0010	.0055	17.36	0.9337	0.9427
	25	0.8958	0.0011	.0124	17.30	0.9100	0.9426
3.0	20	0.9711	0.0010	.0071	17.34	0.9798	0.9427
	30	0.9197	0.0010	.0176	17.27	0.9389	0.9425
3.5	35	0.9389	0.0010	.0146	17.24	0.9551	0.9423
	40	0.9171	0.0010	.0185	17.22	0.9372	0.9423
	45	0.8997	0.0010	.0224	17.20	0.9237	0.9422
4.0	45	0.9357	0.0009	.0224	17.18	0.9596	0.9422
	50	0.9137	0.0008	.0263	17.18	0.9413	0.9421

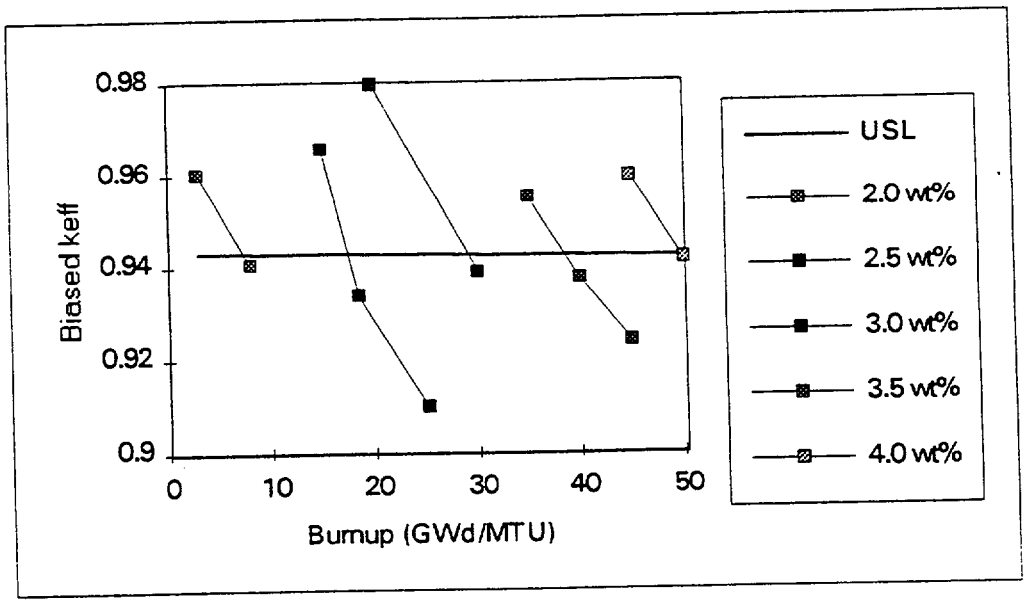
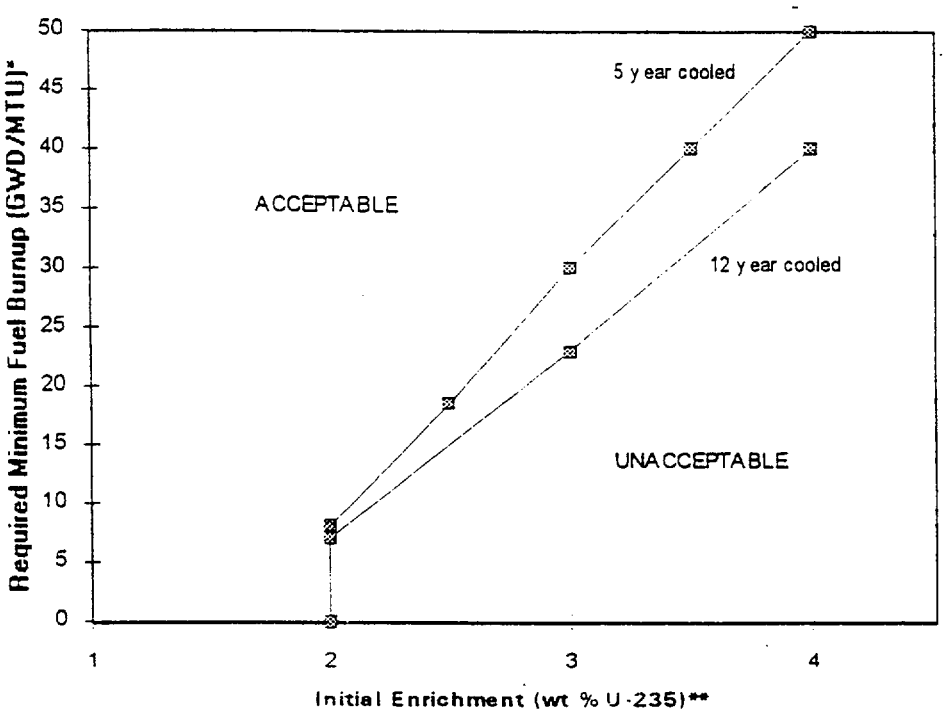


Figure A.2-3. Determination of Required Minimum Burnups



Assembly Design: Westinghouse 17x17

Minimum Cooling Time: 5 years

Maximum Number of Removable Burnable Poison Rods: 0

Note: This loading curve was generated with the following assumptions: Maximum Cycle Average Boron of 650 ppm, Maximum Core Outlet Temperature of 570 K, and Maximum Pellet Average Temperature of 900 K.

* 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 A.2-4. MPC-32 Loading Curve using Actinide-Only Burnup Credit

A.3.1 TN-40 PACKAGE MODEL AND FUEL TYPE

The containment vessel for the TN-40 cask consists of: an inner shell which is a welded, carbon steel cylinder with an integrally-welded, carbon steel bottom closure; a welded flange forging; a flanged and bolted carbon steel lid with bolts; and penetration assemblies with bolts. The overall containment vessel length is 175.0 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a diameter of 72.0 in. and a length of 163.0 in..

There are two penetrations through the containment vessel, both in the lid: one is for a drain opening and the other is for venting. A double-seal mechanical closure is provided for each penetration. The containment lid is 4.50 in. thick and is fastened to the body by 48 bolts.

A gamma shield is provided around the walls of the containment vessel by an independent shell of carbon steel which is welded to a bottom shield plate and to the closure flange. The gamma shield completely encloses the containment vessel inner shell and bottom closure.

Neutron shielding is provided by a resin compound surrounding the body. The resin compound is enclosed in long, slender aluminum containers. The array of resin-filled containers is enclosed within a smooth outer steel shell constructed of two half cylinders.

The basket structure consists of an assembly of stainless steel cells joined by a proprietary fusion welding process and separated by aluminum and poison plates which form a sandwich panel. The panel consists of two 0.25 in. thick aluminum plates which sandwich a poison plate 0.075 in. thick. The boron loading of the poison plate is 10 mg/cm^2. The aluminum provides the heat conduction paths from the fuel assemblies to the cask cavity wall. The poison material provides the necessary criticality control. This method of construction forms a very strong honeycomb-like structure of cell liners which provide compartments for 40 fuel assemblies. The open dimension of each cell is 8.05 in. x 8.05 in. which provides a minimum of 1/8 in. clearance around the fuel assemblies. The overall basket length (160 in.) is less than the cask cavity length to allow for thermal expansion and fuel assembly handling.

Burnup credit is being evaluated for the TN-40 in order to evaluate the possibility of transporting a loaded TN-40 cask from the ISFSI at some future date. The current storage license allows boron credit to be utilized for criticality control. The KENO calculation model assumes a completely flooded cask cavity with the cask body water reflected all around. The fuel rod plenum and assembly end fittings are modeled as water. Figure A.3.-1 shows a radial portion of the model and indicates locations of the 40 fuel assemblies. The lighter and darker diagonals of the fuel assemblies represents the variation in burnup/reactivity due to assembly "tilt" considerations, with the light color representing the lower burnup fuel pins within the assembly. Table A.3-1 gives the radial dimensions of the cask body. The Westinghouse 14x14 OFA fuel assembly is modeled discretely. Figure A.3-2 shows the center of the basket with the fuel assemblies consisting of the fuel rods and guide tubes filled with water. Table A.3-2 provides the fuel assembly parameters. Table A.3-3 gives the material compositions used in the calculations. Although the Westinghouse 14x14 assembly is physically modeled, the isotopics used for a 17x17 assembly are still valid because the atom density would be essentially the same for both. The ORNL SCALE 4.3 code package was utilized to perform the criticality calculations presented in this appendix. The SCALE package is available from the Radiation and Shielding Information Center (RSIC) at Oak Ridge.

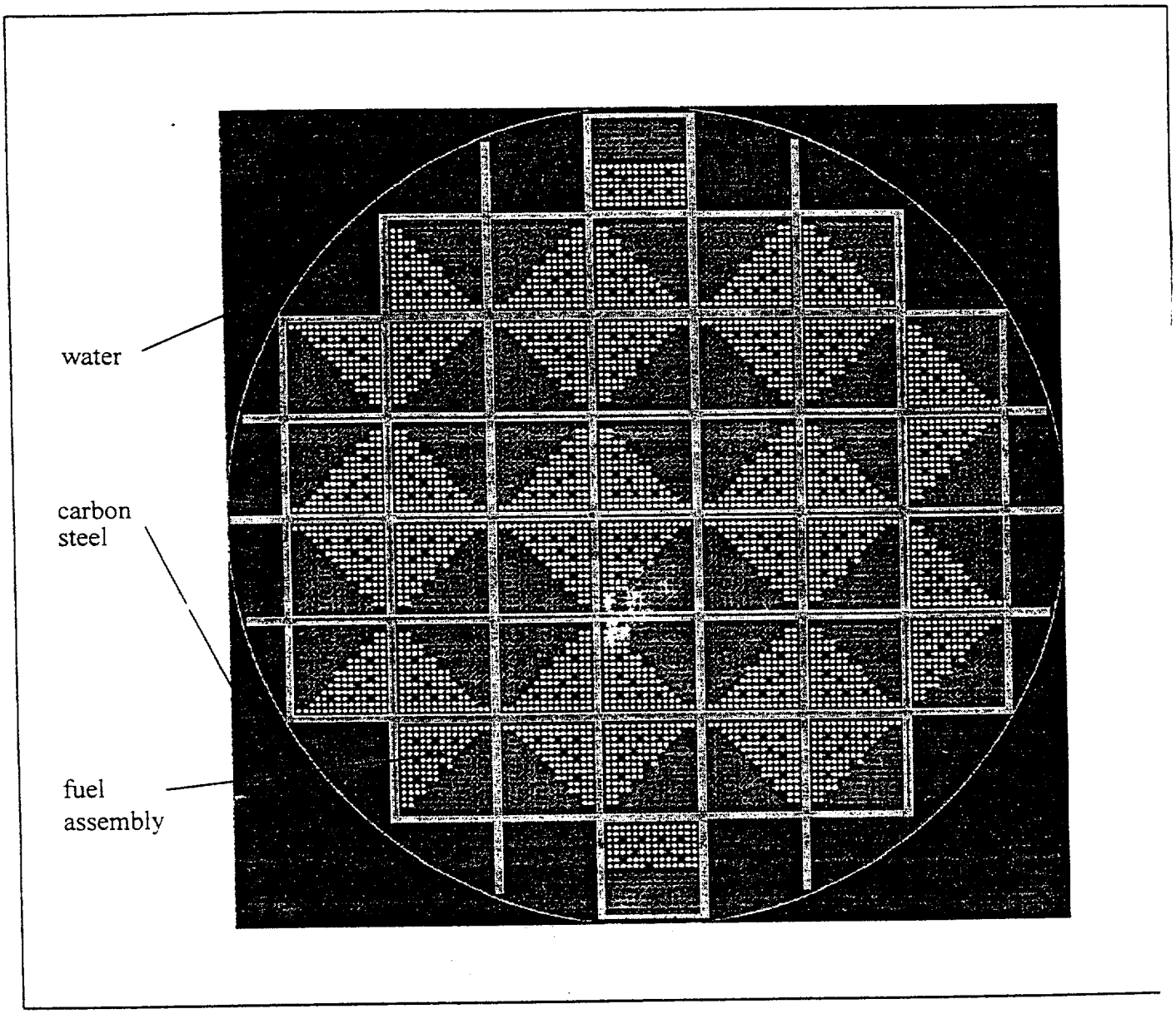


Figure A.3-1 Cross Section of the Keno Model

Table 3A-1 TN-40 Cask Body Radial Dimensions

Material	Outer Radius (cm)
SS304, Al & Boral (basket)	90.8
Water Gap	91.4
Carbon Steel	115.6
Polyester Resin/Al*	127.0
Carbon Steel*	128.3

*- Replaced by water in KENO model.

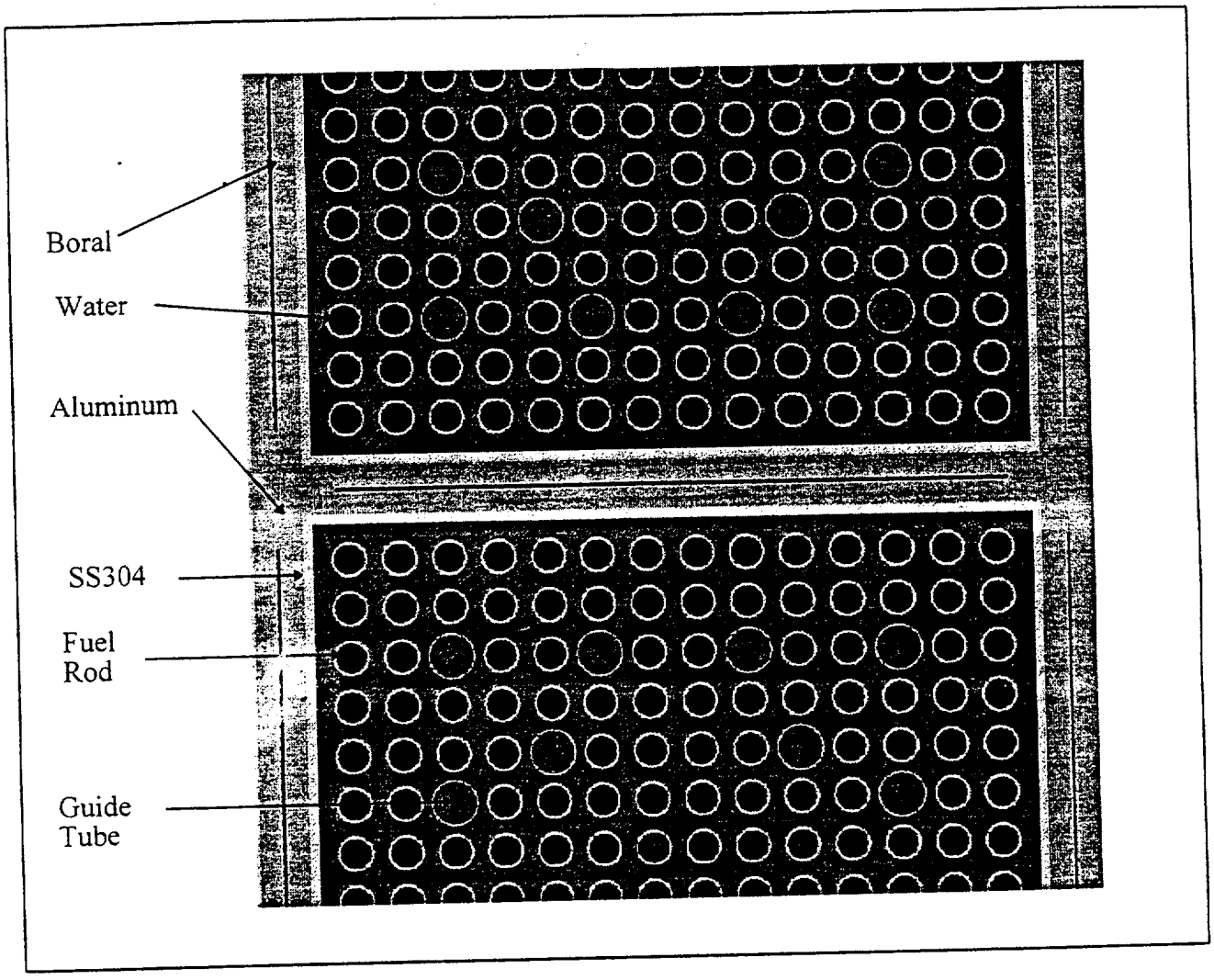


Figure A.3-2 Interior Fuel Assembly Basket with 14x14 Assembly

Table 3.A-2 Westinghouse 14x14 Fuel Assembly Model Parameters

Description	Value
Number of Fuel Rods	179
Number of Water-Filled Guide/Instrument Tubes	17
Active Fuel Length	365.8 cm
Effective Fuel Packing Density	10.39 g/cc
Fuel Rod Pitch	1.412 cm
Clad Material	Zircaloy
Fuel Outside Radius	0.4374 cm
Clad Inside Radius	0.4463 cm
Clad Outside Radius	0.508
Guide Tube Material	Zircaloy
Guide Tube Inside Radius	0.6223 cm
Guide Tube Outside Radius	0.6706 cm

Table 3.A-3 Material Compositions

Material	Density	Element	Atom Density
Zircaloy	6.44	*	4.251E-2
Water	0.988	H	6.6759E-2
		O	3.3380E-2
Stainless Steel	7.92	Cr	1.7430E-2
		Mn	1.7364E-3
		Fe	5.9359E-2
		Ni	7.7182E-3
Carbon Steel	7.82	C	3.9217E-3
		Fe	8.3500E-2
Aluminum	2.699	Al	6.0242E-2
Boral Core	2.63	B10	9.4855E-3
		B11	3.8518E-2
		C	1.2001E-2
		Al	3.4804E-2

* - Zircaloy is a composite cross section for 97.91% Zr, 1.59% Sn, 0.5% Fe.

A.3.2 ISOTOPIC VALIDATION

The SCALE computer code with the 27BURNULIB was utilized for the sample calculations; therefore, isotopic validation was not performed nor necessary.

A.3.3 CRITICALITY VALIDATION

The SCALE computer code with the 27BURNULIB was utilized for the sample calculations; therefore, criticality validation was not performed nor necessary.

A.3.4 LIMITING PARAMETERS

Chapter 4 discusses many of the limiting parameters required for input to the SAS2H isotopic concentration generation process.

A.3.5 CONSTRUCTION OF LOADING CURVE

A.3.5.1 Fresh Fuel Intercept

The fresh fuel calculations are a series of CSAS25 (KENO) calculations using the cask model to determine the maximum fresh fuel enrichment that can be safely loaded into the cask. The fresh fuel enrichments that were evaluated ranged from 1.6 wt% U-235 to 2.2 wt%. Fresh fuel isotopics are shown in Table A.3-4. Table A.3-5 shows the results of the calculations and the Upper Safety Limit (USL) value. The k_{eff} results plus 1.645σ are plotted against the enrichment in Figure A.3-3. The intercept of k_{eff}+1.645σ and the USL is at an enrichment of 1.95 wt% U-235. As a result, 1.95 wt% is the maximum allowable enrichment for fresh Westinghouse 14x14 fuel for loading in the TN-40 cask.

A.3.5.2 Spent Fuel Composition

The spent fuel isotopic composition is determined by executing the code sequence, SAS2H. The isotopics used for these sample calculations are linear interpolations between enrichment and burnup set provided by DOE. The interpolations produced isotopics for 5 yr and for 15 yr cooled fuel. These isotopics are for a standard Westinghouse 17x17 fuel assembly . However, the atom densities calculated here are very similar to those that would be calculated for a 14x14 assembly and are appropriate for the sample calculations included herein.

Table A.3-4 Fresh Fuel Isotopics in Weight Percent

U-235	U-234	U-236	U-238
1.6	0.01290	0.00730	98.3798
1.8	0.01462	0.00828	98.1771
2.0	0.01639	0.00920	97.9744
2.1	0.01728	0.00966	97.8731
2.2	0.01817	0.01012	97.7717

Table A.3-5 Fresh Fuel Calculational Results

Enrichment %	k	σ	$k_{eff} + 1.6\sigma$	USL
1.6	0.8714	0.0015	0.8739	0.9388
1.8	0.9098	0.0015	0.9123	0.9388
2.0	0.9438	0.0016	0.9464	0.9388
2.1	0.9618	0.0017	0.9646	0.9388
2.2	0.9721	0.0016	0.9747	0.9388

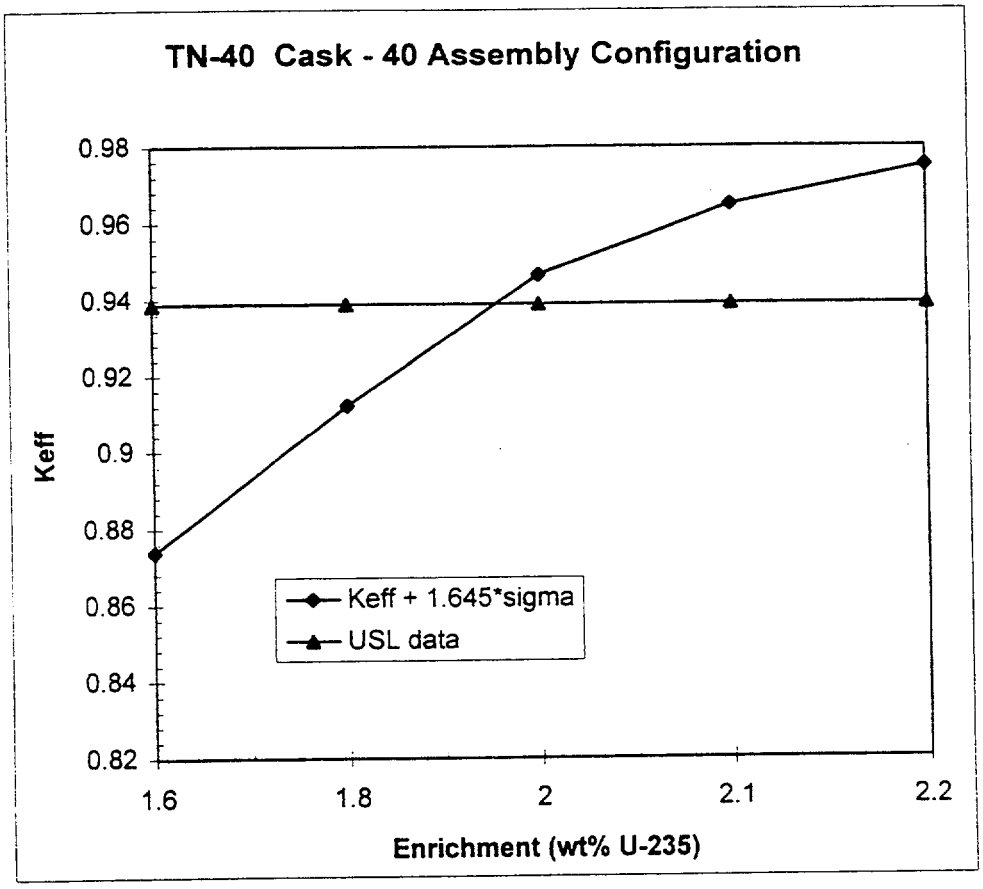


Figure 3A-3 Fresh Fuel Loading Results Considering Biases and Uncertainties

A.3.5.3 Spent Fuel Criticality Calculations

For a specific initial enrichment and burnup, the 15 yr isotopics provided by DOE were used to perform a series of CSAS25 calculations to determine the reactivity of the TN-40 cask loaded with 40 fuel assemblies. Horizontal tilt (variation in burnup) was accounted for in the analyses. The burnups for the assemblies were increased and decreased by predetermined percentages based on the range of the nominal burnup. The burnup ranges were 0-20 GWD/MTU, 20-40 GWD/MTU, and 40-60 GWD/MTU, with the corresponding percentages of ±33%, ±20%, and ±15% respectively. The assemblies were modeled with the orientation as shown in Figure A.3-1.

Using the average lethargy for absorption (ALA) from each calculation, the USL is determined. Table A.3-6 shows the results for the calculated enrichment and burnup pairs that went into the loading curve determination. Figure A.3-4 plots the calculated $k_{eff} + 1.6\sigma$ from the CSAS25 calculations and the evaluated USL for each case. End effects ranging from 0.276% for 25 GWD/MTU burnup to 1.656% for 50 GWD/MTU burnup are also included in the k_{eff} values listed and plotted. Table A.3-7 shows the final interpolated burnup values at the USL intercepts. Figure A.3-5 plots the final required burnup versus initial enrichment. Spent fuel assemblies with minimum burnup and initial enrichment in the region above the loading curve can be loaded safely into the TN-40 cask.

Table A.3-6 Spent Fuel Calculational Results - Actinides Only

Enrichment	Burnup	K_{eff}	σ	ALA	end effct%	K_{eff}*	USL
1.95%	3	0.9529	0.0013	17.85		0.9550	0.9388
	5	0.9468	0.0013	17.81		0.9489	0.9388
	10	0.9201	0.0013	17.78		0.9222	0.9388
	15	0.8877	0.0012	17.75		0.8897	0.9388
2.30%	10	0.9615	0.0012	17.75		0.9635	0.9388
	15	0.9301	0.0012	17.72		0.9321	0.9388
	20	0.8960	0.0012	17.72		0.8980	0.9388
	25	0.8640	0.0012	17.71	0.276	0.8684	0.9369
2.80%	20	0.9485	0.0013	17.67		0.9506	0.9367
	25	0.9165	0.0012	17.67	0.276	0.9210	0.9367
	30	0.8847	0.0012	17.66	0.552	0.8916	0.9367
	35	0.8705	0.0012	17.7	0.828	0.8797	0.9369
3.30%	25	0.9648	0.0012	17.62	0.276	0.9694	0.9365
	30	0.9320	0.0013	17.63	0.552	0.9393	0.9366
	35	0.9187	0.0012	17.65	0.828	0.9283	0.9367
	40	0.8884	0.0011	17.67	1.104	0.9000	0.9367
	45	0.8669	0.0012	17.67	1.380	0.8808	0.9367
3.85%	40	0.9363	0.0012	17.63	1.104	0.9486	0.9366
	45	0.9093	0.0012	17.64	1.380	0.9238	0.9366
	50	0.8957	0.0012	17.67	1.656	0.9125	0.9368

* - Includes 1.645σ and end effects

Figure A.3-4 Calculational Results Leading to Loading Curve Generation

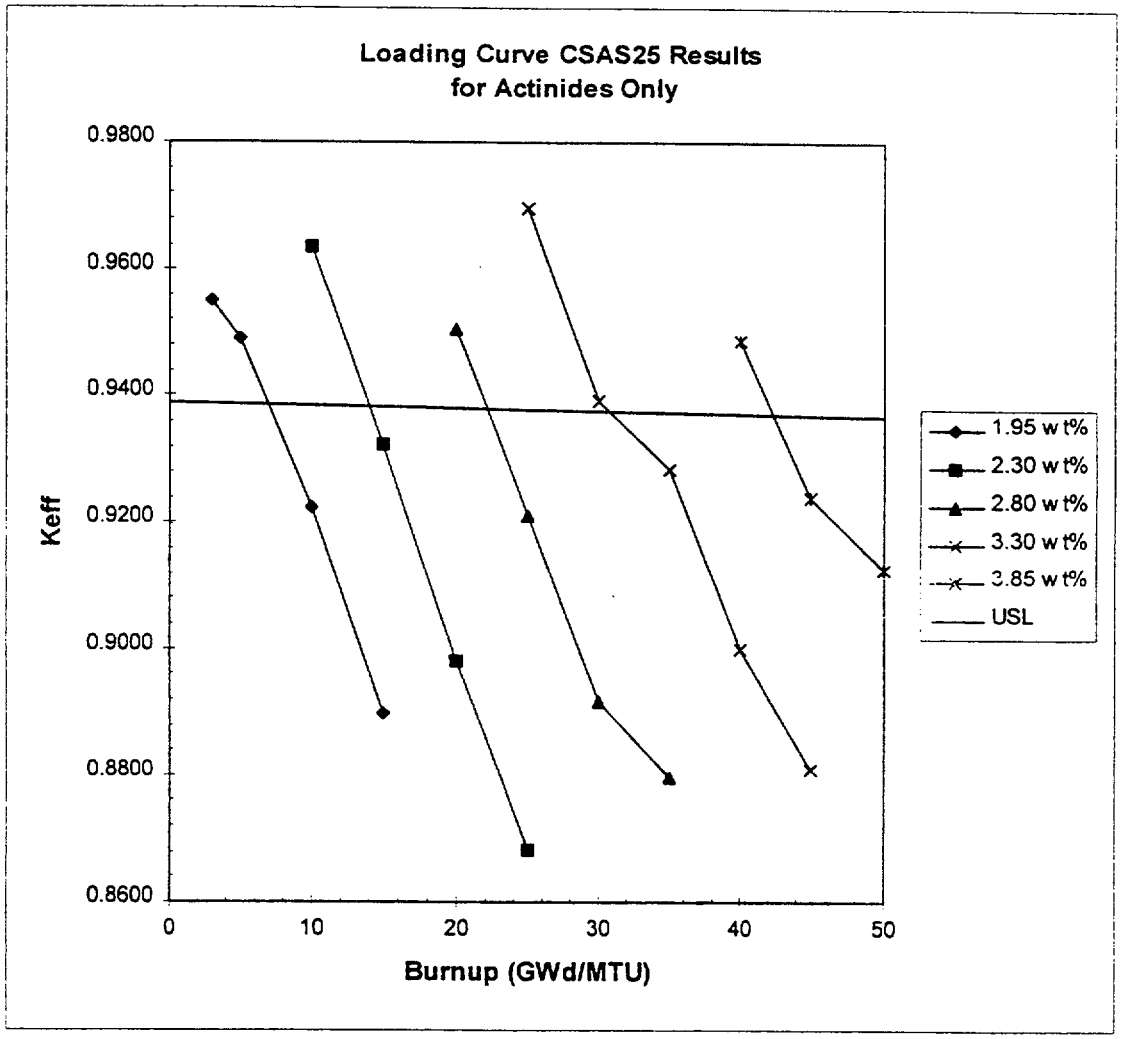
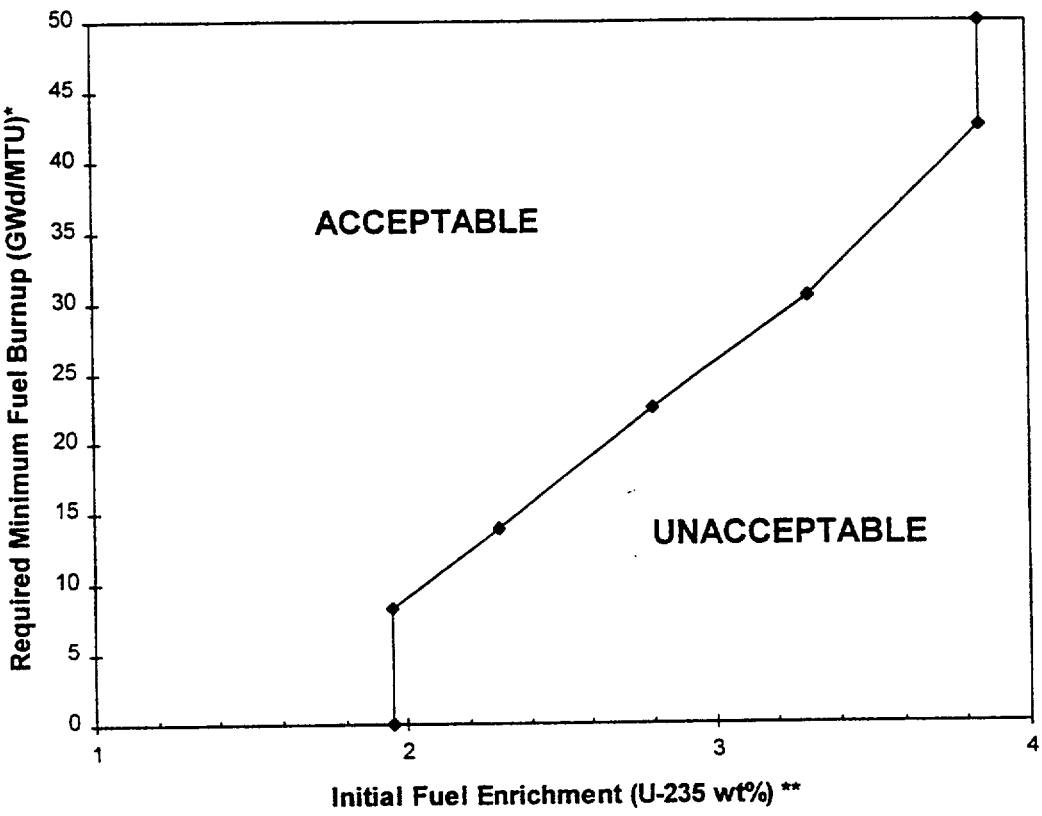


Table A.3-7 TN-40 Loading Curve Data for Westinghouse 14 x 14 Fuel (Actinides Only)

Initial Enrichment (wt%)	Burnup (GWD/MTU)	USL
1.95	0 (fresh fuel)	0.9388
1.95	8.32	0.9388
2.3	13.92	0.9388
2.8	22.00	0.9367
3.3	30.43	0.9367
3.85	42.42	0.9366



Assembly Design: W 14x14

Minimum Cooling Time: 15 yr

Maximum Number of Removable Burnable Absorber Rods: 0

Note: This loading curve was generated with the following generic assumptions: Maximum Cycle Average ppm Boron of 650, Maximum Core Outlet Temperature of 570°K, and the Maximum Pellet Average Temperature of 900°K.

* 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 A.3-5 Development of Burnup Credit Loading Curve for the TN-40 Spent Fuel Cask

A.4 NUHOMS®-MP187 SAMPLE BURNUP CREDIT ANALYSIS

This Appendix demonstrates the application of the methodology described in Chapters 2 through 7 to VECTRA's NUHOMS®-MP187 multi-purpose cask. This demonstration will be accomplished by developing a sample loading curve for the cask as described below.

A.4.1 CASK DESCRIPTION

The NUHOMS®-MP187 cask is the transportation overpack for the NUHOMS® Multi-Purpose Canister (MPC) system. The cask, along with three PWR spent fuel canisters, was submitted to the U.S. NRC for storage and transportation licensing in September 1993. Details of the cask and canisters can be found in, "Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask," Docket 71-9255. The cask itself, shown in Figure A.4-1, provides an overpack for the fuel canisters during onsite transfer, onsite storage (as an alternate to the NUHOMS® concrete horizontal storage modules), and offsite transportation. The cask body consists of stainless steel inner and outer shells, stainless steel top and bottom forgings, a lead gamma shield layer, and a Bisco NS-3 neutron shield layer. An opening in the cask bottom end allows fuel canisters to be handled either horizontally or vertically. The cask exterior has a length of 201.5 inches and a diameter of 92.5 inches. The cask cavity length is 187 inches and the cavity diameter is 68 inches.

The 24 element PWR fuel canister is shown in Figure A.4-2. The basket assembly consists of steel spacer discs, support rods, and guide sleeves. Each guide sleeve includes neutron absorber sheets which form flux traps between adjacent assemblies. The spacer discs and support rods serve to maintain the gaps between the fuel assemblies. The canister basket is surrounded by a welded stainless steel shell which includes shielding at both ends to minimize occupational exposures during handling. Each fuel "cell" has an open width of 8.9 inches and a length of either 167 inches or 173 inches depending on the canister type.

The current license applications for the MP187 cask are based on a fresh fuel enrichment of 3.43 wt% U-235. While this enrichment bounds most of the fuel currently stored in plant fuel pools, it will not bound a large portion of the fuel being discharged either currently or in the future. As shown in the remainder of this Appendix, applying a burnup credit analysis to the existing cask/canister designs will easily allow fuel with maximum initial enrichments of up to 5 wt% U-235 to be stored/transported in the MP187 cask. Future work will include removing absorber sheets to reduce the canister costs and new canister designs with increased payloads.

This sample analysis was performed using the ORNL SCALE 4.3 code package, available from the Radiation Shielding Information Center, installed on a personal computer. With the exception of the fuel assemblies, the KENO analytical model is identical to that described in Chapter 6 of Revision 3 of the MP187 SAR. Parameters identical to those in the MP187 SAR include the package geometry (including worst case tolerances) and material compositions. For

consistency with the isotopic data obtained from the DOE, a Westinghouse 17x17 fuel assembly is used in this sample calculation which differs from the B&W 15x15 assembly which represents the design basis for the MP187 package. Table A.4-1 provides the Westinghouse 17x17 fuel specifications.

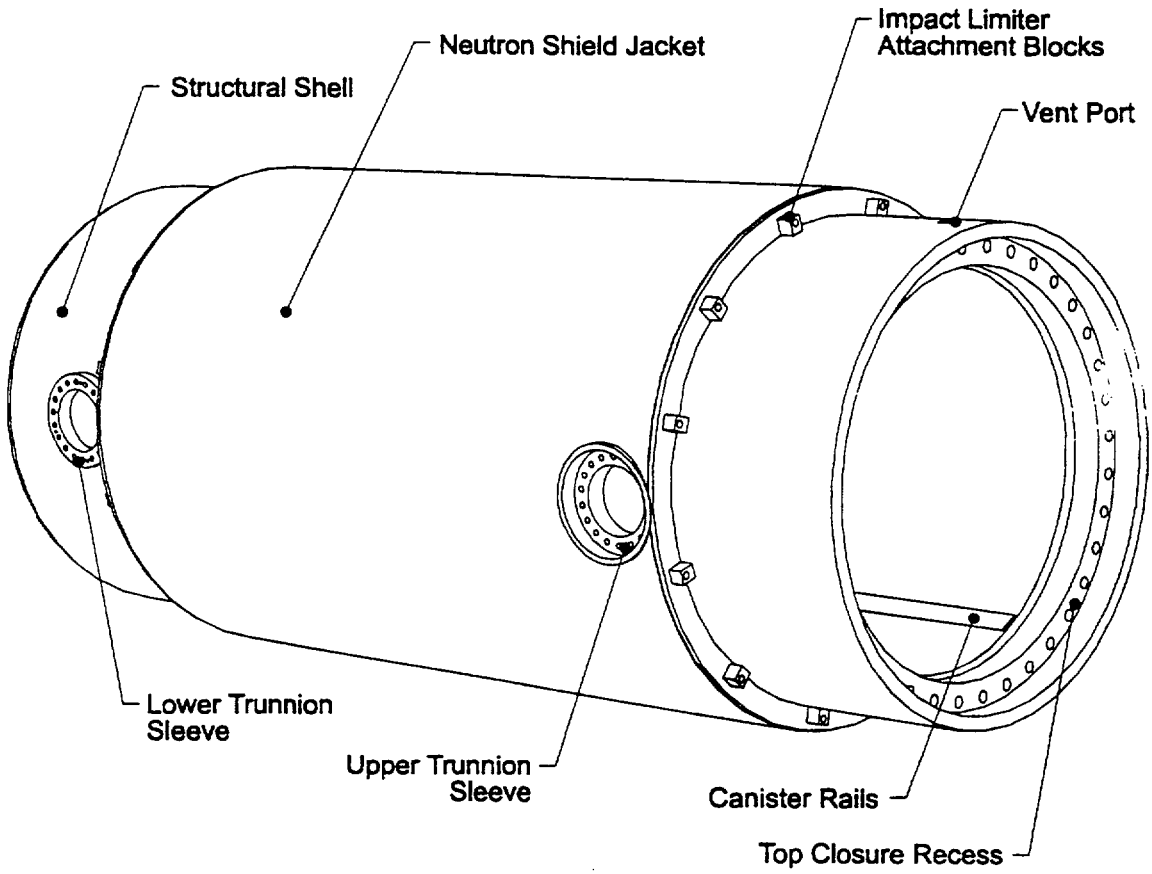


Figure A.4-1: NUHOMS®-MP187 Multi-Purpose Cask

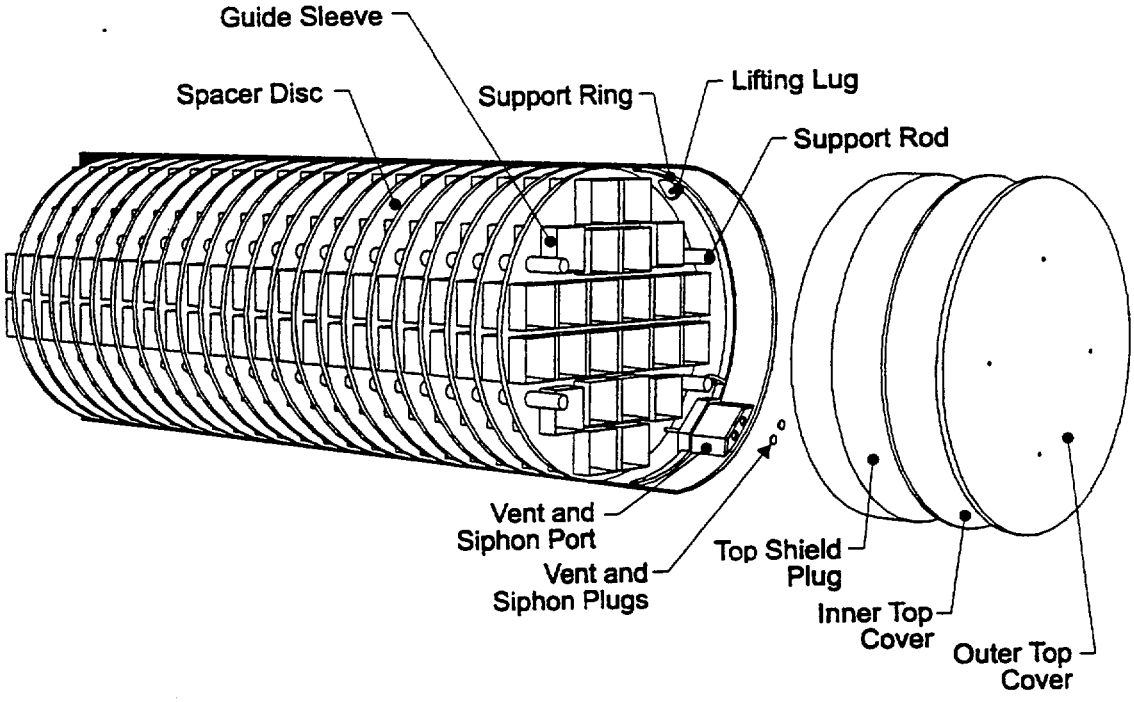


Figure A.4-2: NUHOMS® 24 Element PWR Canister

Table A.4-1: Westinghouse 17x17 Fuel Assembly Model Parameters

Description	Value
Number of Fuel Rod Positions	289
Number of Fueled Rods	264
Number of Water Filled Guide/Instrument Tubes	25
Number of Burnable Absorber/Control Rods	0
Active Fuel Length	144 in
Fuel Material	UO_2
Fuel Rod Pitch	0.496 in
Clad Material	Zircaloy-4
Rod Diameter	0.374 in
Clad Thickness	0.0225 in
Fuel Diameter	0.3225 in
Guide/Instrument Tube Material	Zircaloy-4

A.4.2 ISOTOPIC VALIDATION

This evaluation was performed using the SCALE criticality sequences with the 27BURNUPLIB cross-section library. Because this code and library are specifically addressed in the body of this report, no additional isotopic validation is required for this sample calculation.

A.4.3 CRITICALITY VALIDATION

This evaluation was performed using the SCALE criticality sequences with the 27BURNUPLIB cross-section library. Because this code and library are specifically addressed in the body of this report, no additional criticality validation is required.

A.4.4 LIMITING PARAMETERS

Chapter 4 discusses many of the limiting parameters required for input to the SAS2H isotopic concentration generation process. Table A.4-2 provides the typical limiting values for the Westinghouse 17x17 assembly used in the SAS2H model. Other factors or uncertainties such as material and fabrication tolerance, modeling limitations, and clustering of assemblies in baskets that are not unique to burnup credit design were not considered in this sample calculation.

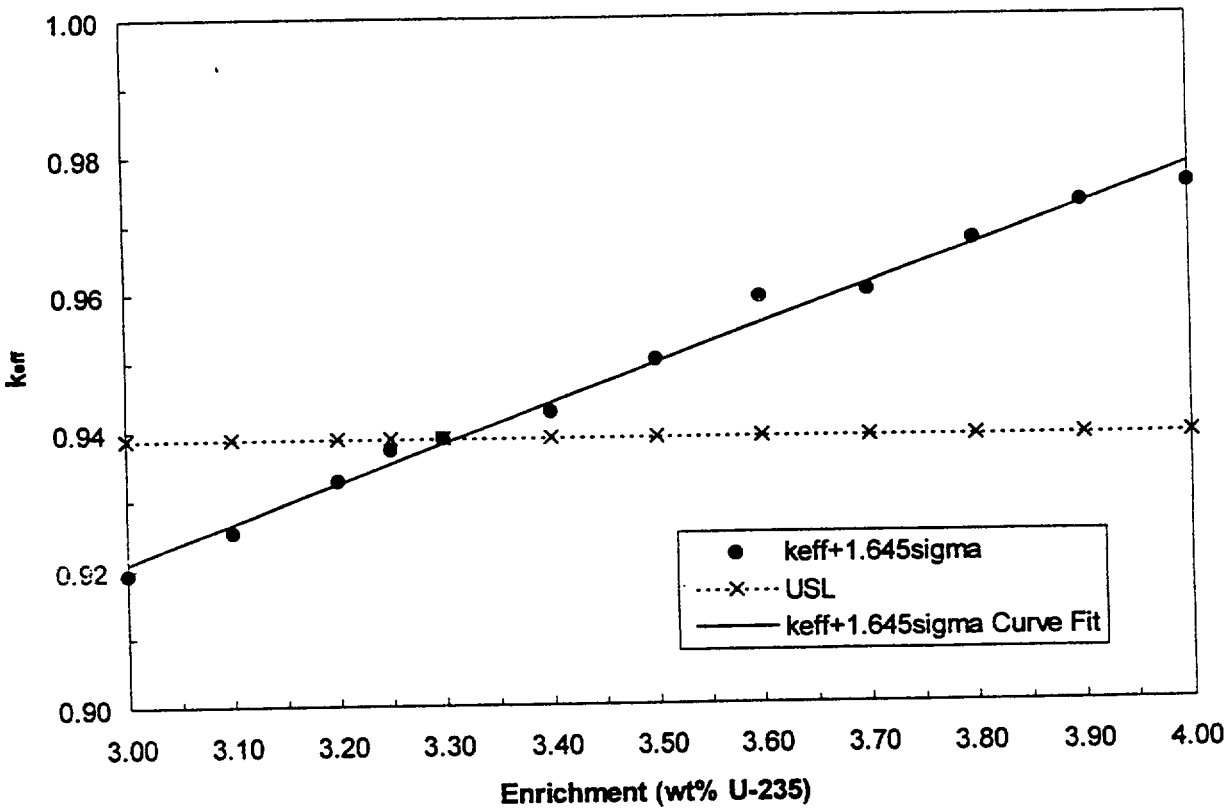
Table A.4-2: Fuel History Parameters for the SAS2H Calculations

Description	Value
Specific Power	60 MW/MTU
Operating History	Single Irradiation Cycle
Boron Concentration	650 ppm
Max. Core Outlet Temp.	570 K
Max. Assembly Avg. Pellet Temp.	900 K

A.4.5 CONSTRUCTION OF LOADING CURVE

A.4.5.1 Fresh Fuel Intercept

The fresh fuel intercept was calculated using a series of CSAS25 (KENO Va) runs and the MP187 cask model. This intercept represents the maximum fresh fuel enrichment that can be safely loaded into the cask. The fuel data presented in Table A.4-1 was used to generate the fresh fuel isotopics as a function of enrichment. The k_{eff} results plus 1.645σ were then plotted against the Upper Safety Limit (USL) as shown in Figure A.4-3. The fresh fuel (uranium) USL is 0.9388 as discussed in Chapter 3. The intercept of $k_{eff} + 1.645\sigma$ and the USL falls between 3.25 wt% and 3.3 wt%. A maximum allowable enrichment of 3.25 wt% U-235 is, therefore, conservatively used for fresh Westinghouse 17x17 fuel in the 24 element MP187 package. Note that this fresh fuel enrichment is less than the 3.43 wt% enrichment for which the cask is currently being licensed. This is due primarily to the use of a USL of 0.95 in the licensing calculations versus 0.9388 in the burnup credit calculations. A sample CSAS25 input file is provided in Table A.4-3



Enrichment (%)	k_{eff}	σ	$k_{eff}+1.645\sigma$	USL
3.00	0.91692	0.00141	0.9192	0.9388
3.10	0.92285	0.00146	0.9253	0.9388
3.20	0.93037	0.00145	0.9328	0.9388
3.25	0.93480	0.00148	0.9372	0.9388
3.30	0.93657	0.00144	0.9389	0.9388
3.40	0.94017	0.00149	0.9426	0.9388
3.50	0.94760	0.00147	0.9500	0.9388
3.60	0.95662	0.00147	0.9590	0.9388
3.70	0.95759	0.00150	0.9601	0.9388
3.80	0.96499	0.00153	0.9675	0.9388
3.90	0.97040	0.00140	0.9727	0.9388
4.00	0.97279	0.00149	0.9752	0.9388

Figure A.4-3: Fresh Fuel Loading Results Considering Biases and Uncertainties

Table A.4-3: CSAS25 Input File for Fresh Fuel Case

```
=csas25
bc325.ki, mp-187 cask, fo/fc dsc 24P, W 17x17 LOPAR, 2/97
27groupndf4 latticecell
uo2      1 0.95 293 92235 3.25 92238 96.75 end
cr       2 0 7.5166-5  293 end
fe       2 0 1.4696-4  293 end
ni       2 0 2.3299-6  293 end
zr       2 0 4.2711-2  293 end
h2o      3   1.0       293 end
fe       4 0 8.3801-2  293 end
mn       4 0 8.6048-4  293 end
cr       5 0 1.7274-2  293 end
mn       5 0 1.7210-3  293 end
fe       5 0 5.9042-2  293 end
ni       5 0 7.4481-3  293 end
al       6 0 3.9268-2  293 end
b-10     6 0 4.8788-3  293 end
b-11     6 0 2.0000-2  293 end
c        6 0 7.6705-3  293 end
pb       7 0 3.2960-2  293 end
al       8 0 7.0275-3  293 end
h        8 0 5.0996-2  293 end
si       8 0 1.2680-3  293 end
ca       8 0 1.4835-3  293 end
fe       8 0 1.0628-4  293 end
c        8 0 8.2505-3  293 end
o        8 0 3.7793-2  293 end
h2o      9   0.3       293 end
h2o     10   1.0       293 end
h2o     11   1.0       293 end
cr      12 0 7.5166-5  293 end
fe      12 0 1.4696-4  293 end
ni      12 0 2.3299-6  293 end
zr      12 0 4.2711-2  293 end
end comp
squarepitch 1.25984 0.81915 1 3 0.94996 2  0.83566 10  end
more data dab=400  end
bc4.ki, mp-187 cask, fo/fc dsc 24P, W 17x17 LOPAR, 2/97
read para
   tme=300     run=yes     far=no  flx=no   fdn=no
   gen=615     npg=400     nsk=15  plt=no   lng=500000
end para
read geom
                        ***** Proprietary geometry data deleted *****
end geom
read array
   com='W 17x17 LOPAR fuel assembly slice, sd'
   ara=1       nux=17      nuy=17      nuz=1
   loop
        29         1 17  1     1 17  1       1 1 1
        30         3 15 12     6 12  3       1 1 1
        30         4 14 10     4 14 10       1 1 1
        30         6 12  3     3 15  3       1 1 1
        31         9  9  1     9  9  1       1 1 1
   end loop
   com='W 17x17 LOPAR fuel assembly slice, f (5.5")'
   ara=2       nux=17      nuy=17      nuz=1
   loop
        61         1 17  1     1 17  1       1 1 1
        62         3 15 12     6 12  3       1 1 1
        62         4 14 10     4 14 10       1 1 1
        62         6 12  3     3 15  3       1 1 1
        63         9  9  1     9  9  1       1 1 1
   end loop
   com='W 17x17 LOPAR fuel assembly slice, f (6.0")'
   ara=3       nux=17      nuy=17      nuz=1
   loop
```

```
      94        1 17  1    1 17  1       1 1 1
      95        3 15 12    6 12  3       1 1 1
      95        4 14 10    4 14 10       1 1 1
      95        6 12  3    3 15  3       1 1 1
      96        9  9  1    9  9  1       1 1 1
   end loop
   com='W 17x17 LOPAR fuel assembly slice, f (6.5")'
   ara=4       nux=17      nuy=17      nuz=1
   loop
      127       1 17  1    1 17  1       1 1 1
      128       3 15 12    6 12  3       1 1 1
      128       4 14 10    4 14 10       1 1 1
      128       6 12  3    3 15  3       1 1 1
      129       9  9  1    9  9  1       1 1 1
   end loop
   com='W 17x17 LOPAR fuel assembly slice, f (6.75")'
   ara=5       nux=17      nuy=17      nuz=1
   loop
      160       1 17  1    1 17  1       1 1 1
      161       3 15 12    6 12  3       1 1 1
      161       4 14 10    4 14 10       1 1 1
      161       6 12  3    3 15  3       1 1 1
      162       9  9  1    9  9  1       1 1 1
   end loop
   com='W 17x17 LOPAR fuel assembly slice, non f (6.0")'
   ara=6       nux=17      nuy=17      nuz=1
   loop
      200       1 17  1    1 17  1       1 1 1
      201       3 15 12    6 12  3       1 1 1
      201       4 14 10    4 14 10       1 1 1
      201       6 12  3    3 15  3       1 1 1
      202       9  9  1    9  9  1       1 1 1
   end loop
   com='stack disks to approximate dsc basket'
   ara=7       nux=1       nuy=1       nuz=54
   fill      171 170 165 166 165 166 165 166 165 167 165 168
             165 169 165 169 165 169 165 169 165 169 165 169
             165 169 165 169 165 169 165 169 165 169 165 169
             165 168 165 168 165 168 165 168 165 167 165 208
             207 209 207 207 210 211
   end fill
end array
read start
   nst=1
end start
read bounds
   xyf=specular
   zfc=water
end bounds
read plot
        ttl='cask material plot - plan view'
        pic=mat
        nch=' fzmcsblwgwz'
        xul=-132  yul= 132 zul=200
        xlr= 132  ylr=-132 zlr=200
        uax=1.0  vdn=-1.0
        nax=132
        plt=yes
        ttl='FA material plot - plan view'
        pic=mat
        nch=' fzmcsblwgwz'
        xul=0.0   yul= 0.0  zul=205
        xlr=12.0  ylr=-12.0 zlr=205
        uax=1.0  vdn=-1.0
        nax=132
        plt=no
end plot
end data
end
```

A.4.5.2 Spent Fuel Composition

Spent fuel compositions for the Westinghouse 17x17 assemblies were calculated as a function of initial enrichment and burnup using linear interpolations of the SAS2H results provided by DOE.

A.4.5.3 Spent Fuel Criticality Calculations

The spent fuel criticality calculations were performed by modifying the fresh fuel CSAS25 models described above to include the fuel isotopics as a function of initial enrichment and burnup. Two burnup regions were defined for each fuel assembly as shown in Figure A.4-4 to account for radial burnup tilt. Although several assembly orientations were analyzed to determine the worst case basket configuration, a more detailed evaluation will be required for a final licensing application. For average fuel burnups less than 20 GWd/MTU, the isotopics in the two fuel regions are based on burnup variations of ± 33%. For average fuel burnups greater than or equal to 20 GWd/MTU, the isotopics are based on burnup variations of ± 20%.

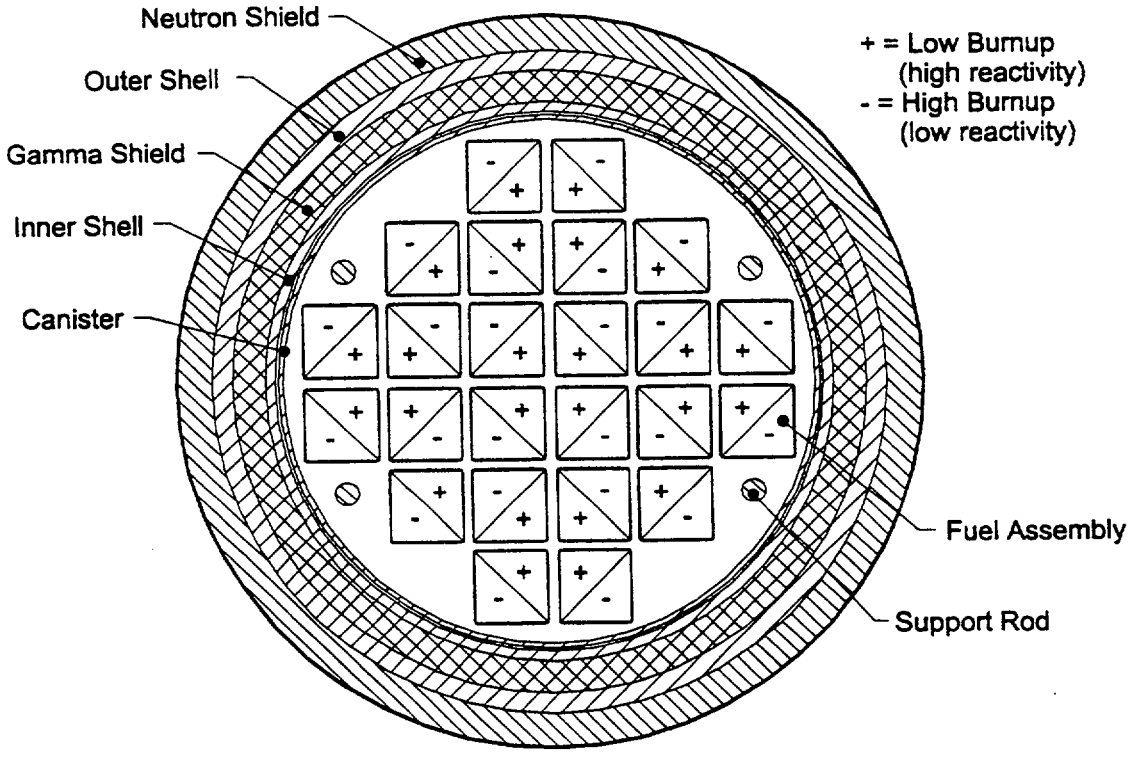


Figure A.4-4: Cross-Section of KENO Model Showing Radial Tilts

Two CSASN models were run for each case to generate the resonance corrected working libraries for the two assembly burnup regions. SCALE's WAX module was then used to combine the working libraries for use by the CSAS25 model, which calculates k_{eff}. The Average Lethargy for Absorption (ALA) was determined for each case from the CSAS25 output and used to calculate the USL as described in Chapter 3.

Table A.4-4 shows the results for the calculated enrichment and burnup pairs that went into the loading curve determination. The enrichments shown in Table A.4-4 are in wt% U-235 and the burnups in GWd/MTU. The values of k_{eff} and σ are taken directly from the CSAS25 output. The final value of k_{eff} which is compared to the USL includes the calculated k_{eff} plus $1.645*\sigma$ plus the end effects bias (k_{bias}). The end effects bias is calculated as discussed in the body of this report. This data is shown graphically on Figure A.4-5.

Figure A.4-6 plots the final required burnup versus initial enrichment. As explained in Chapter 5, any spent fuel assemblies with a burnup and initial enrichment under the loading curve (unacceptable region) will not be qualified for loading into the MP187 cask. Spent fuel assemblies with a burnup and initial enrichment above the curve (acceptable region) can be safely loaded into the cask.

A.4.5.4 Low-Density Moderation Effects

This analysis was not performed for this sample calculation because, as discussed in Chapter 6 of the MP187 SAR, a water density of 1.0 is limiting for the NUHOMS®-MP187 package.

Table A.4-4: Spent Fuel Calculational Results - Actinide Only

Enrichment	Burnup	k_{eff}	σ	ALA	$k_{eff}+1.645\sigma+k_{bias}$	USL
3.25	1	0.94435	0.00120	17.62	0.94632	0.93650
	5	0.92790	0.00120	17.60	0.92987	0.93642
	10	0.90636	0.00124	17.59	0.90840	0.93636
3.50	5	0.94403	0.00131	17.57	0.94618	0.93629
	8	0.93222	0.00118	17.56	0.93416	0.93623
	15	0.89496	0.00120	17.57	0.89693	0.93628
4.00	10	0.94611	0.00123	17.52	0.94813	0.93601
	13	0.93503	0.00118	17.50	0.93697	0.93594
	14	0.92957	0.00124	17.50	0.93161	0.93595
	20	0.90196	0.00116	17.52	0.90387	0.93604
4.50	15	0.95066	0.00123	17.45	0.95268	0.93570
	20	0.92940	0.00121	17.47	0.93139	0.93579
	25	0.91009	0.00116	17.48	0.91476	0.93584
5.00	20	0.95060	0.00115	17.41	0.95249	0.93549
	25	0.93160	0.00125	17.43	0.93642	0.93560
	26	0.92761	0.00120	17.43	0.93290	0.93562
	30	0.90962	0.00122	17.44	0.91715	0.93567

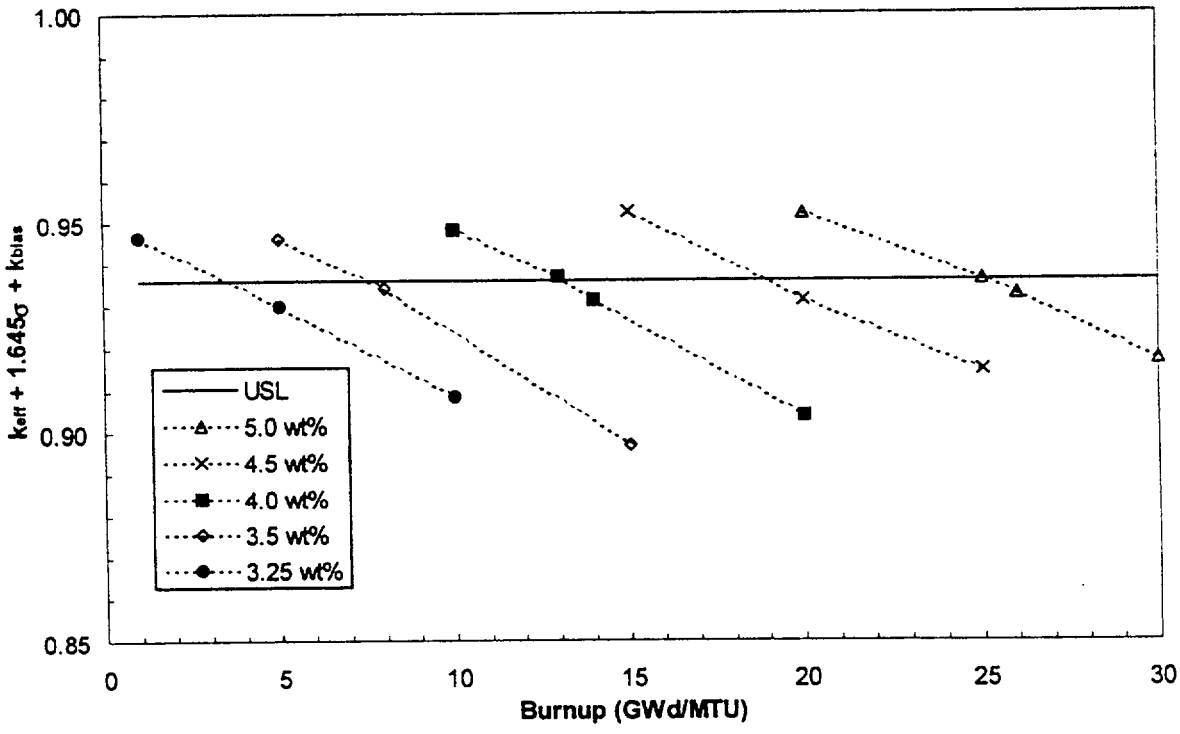
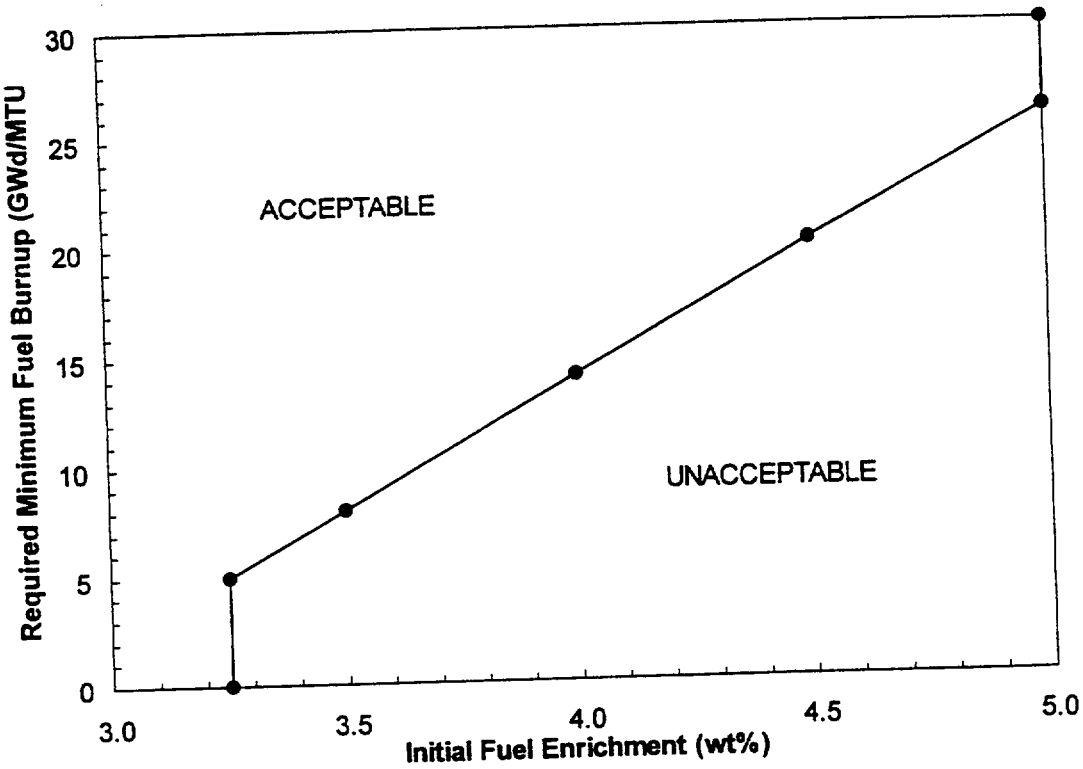


Figure A.4-5: Calculational Results Used to Generate the Loading Curve



Assembly Design: Westinghouse 17x17

Minimum Cooling Time: 5 years

Maximum Number of Removable Burnable Poison Rods: 0

Note: This loading curve was generated with the following assumptions: Maximum Cycle Average Boron of 650 ppm, Maximum Core Outlet Temperature of 570 K, and Maximum Pellet Average Temperature of 900 K.

* 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 A.4-6: Loading Curve for WE 17x17 Fuel Assemblies in the NUHOMS®-MP187 Cask

APPENDIX B

BURNUP MEASUREMENT SYSTEMS

The following are descriptions of burnup measurement systems as provided by the vendor. Appendix B.1 is a description of the BNFL FuelMaster. Appendix B.2 is a description of EPRI's Fork+ system.

BNFL INSTRUMENTS BURNUP MEASUREMENT SYSTEM

B.1.1 Introduction

Spent fuel burnup monitoring is being offered as a service using an instrument system based on high resolution gamma spectrometry (HRGS). The system, known as FuelMaster™ (formerly Spent Fuel Monitor (SFM)), has been designed using experience gained by the development and use of a series of instruments for the support of operations at the Sellafield Nuclear Fuel Reprocessing Facility in the United Kingdom. Industrial robustness with low maintenance and high reliability have been key attributes of these instruments and have allowed the successful measurement of more than 1 million fuel items since the 1970s.

As part of the FuelMaster™ development program, a demonstration system was produced in 1996. Using this system, measurements were made on 55 PWR assemblies in a US utility spent fuel pool during April 1996. The measurements presented the opportunity to provide the utility with their required burnup verification data and to simultaneously demonstrate the BNFL Instruments system to the US Department of Energy (USDOE) and the Nuclear Regulatory Commission (NRC). Subsequently, a commercial contract was awarded by the same utility to measure an additional campaign of more than 300 PWR assemblies. A preliminary report on these measurements has been published under the auspices of the Electric Power Research Institute (EPRI)[1]. The report concluded that HRGS measurements using the FueMaster™ could provide accurate determination of burnup and cooling time. Based on a dependent calibration, the correlation between measured and reactor records burnup was shown to have a 1σ error of between 3 and 5%.

B.1.2 Design

A schematic illustration of the demonstration FuelMaster™ is shown in figure B.1-1. The simplicity of design allows ease of installation, removal and decontamination, and is intended to minimise disruption to other operations in the pool during its use. This configuration is pool wall mounted and fuel assemblies are brought to it for measurement. Movement of fuel to the monitoring system for measurement was considered appropriate as fuel measurements may be co-ordinated with other operations such as fuel inspection or fuel transfer to either dry storage/transport casks or to a segregated area for measured assemblies. The principal mechanical components comprise:

(i) A vertical re-entrant tube that is fixed to the pool wall. The re-entrant tube allows the insertion and removal of the high resolution gamma detector within a carriage. This approach to detector delivery, in which the detector is segregated from the pool, provides direct access to the detector for maintenance and prevents any risk of contamination from the pool water.

(ii) A detector carriage. This acts to protect the detector from mechanical damage and ensures its correct positioning with respect to the gamma collimator axis.

(iii) A shielded detector enclosure. This minimises the magnitude of any background radiation reaching the detector and aids optimisation of gamma spectral quality.

(iv) A horizontal collimator. This defines and controls the detector's field of view at the fuel assembly.

(v) A v-shaped fuel location fixture. This permits simultaneous views of two faces of the assembly and provides a means for accurate and reproducible positioning of the fuel assembly during measurement.

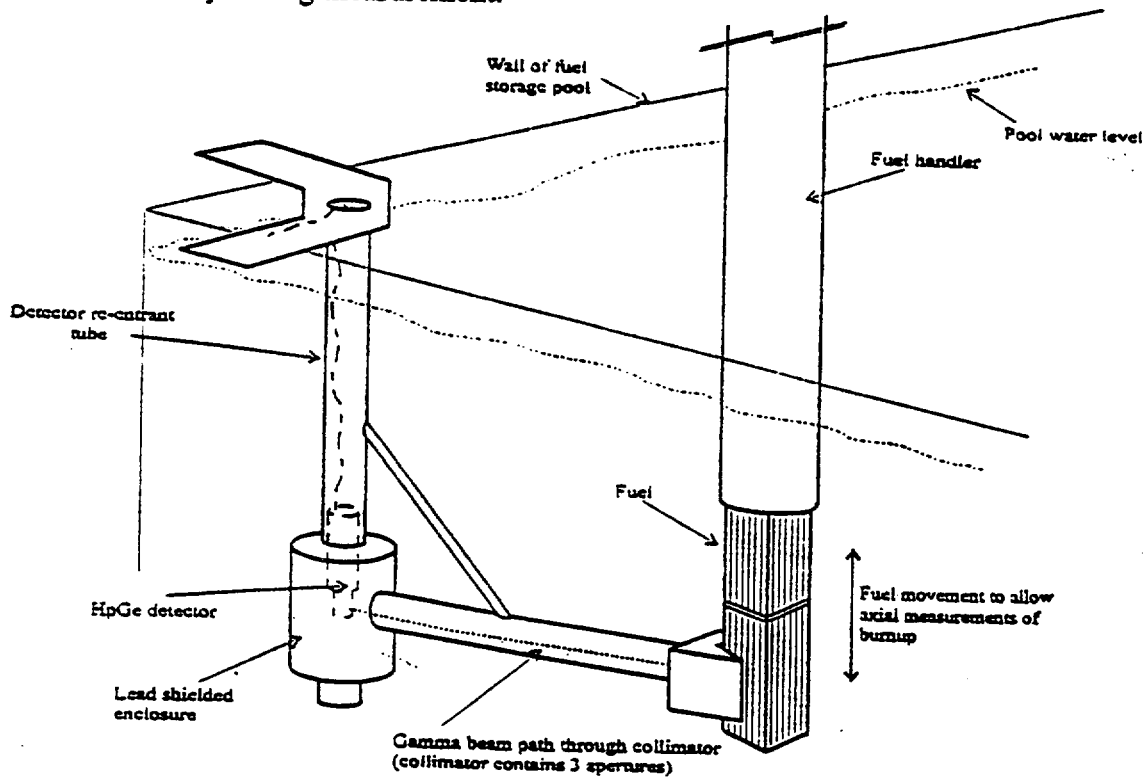


Figure B.1-1 Demonstration FuelMaster™ schematic arrangement.

Photographs of the demonstration system are shown in figures B.1-2 and B.1-3. Figure B.1-2 shows the HRGS high purity intrinsic germanium (HpGe) detector being inserted into the re-entrant tube at the pool side. Figure B.1-3 shows a plan view of the collimator pointing from the pool corner to a fuel assembly held in the fuel handling machine. The fuel is shown as it approaches the v-shaped fixture on the end of the gamma collimator. Both the pond wall fixing plate arrangement, visible in figure B.1-2, and the collimator length may be tailored to suit the local conditions at specific utilities. Adaptations of the FuelMaster™ design could also be used in dry, out of pool, conditions.

All electrical service and signal cables are fed back from the detector to the radiometric, control and data storage electronics through the re-entrant tube. This again allows easy maintenance and eliminates equipment contamination risks.

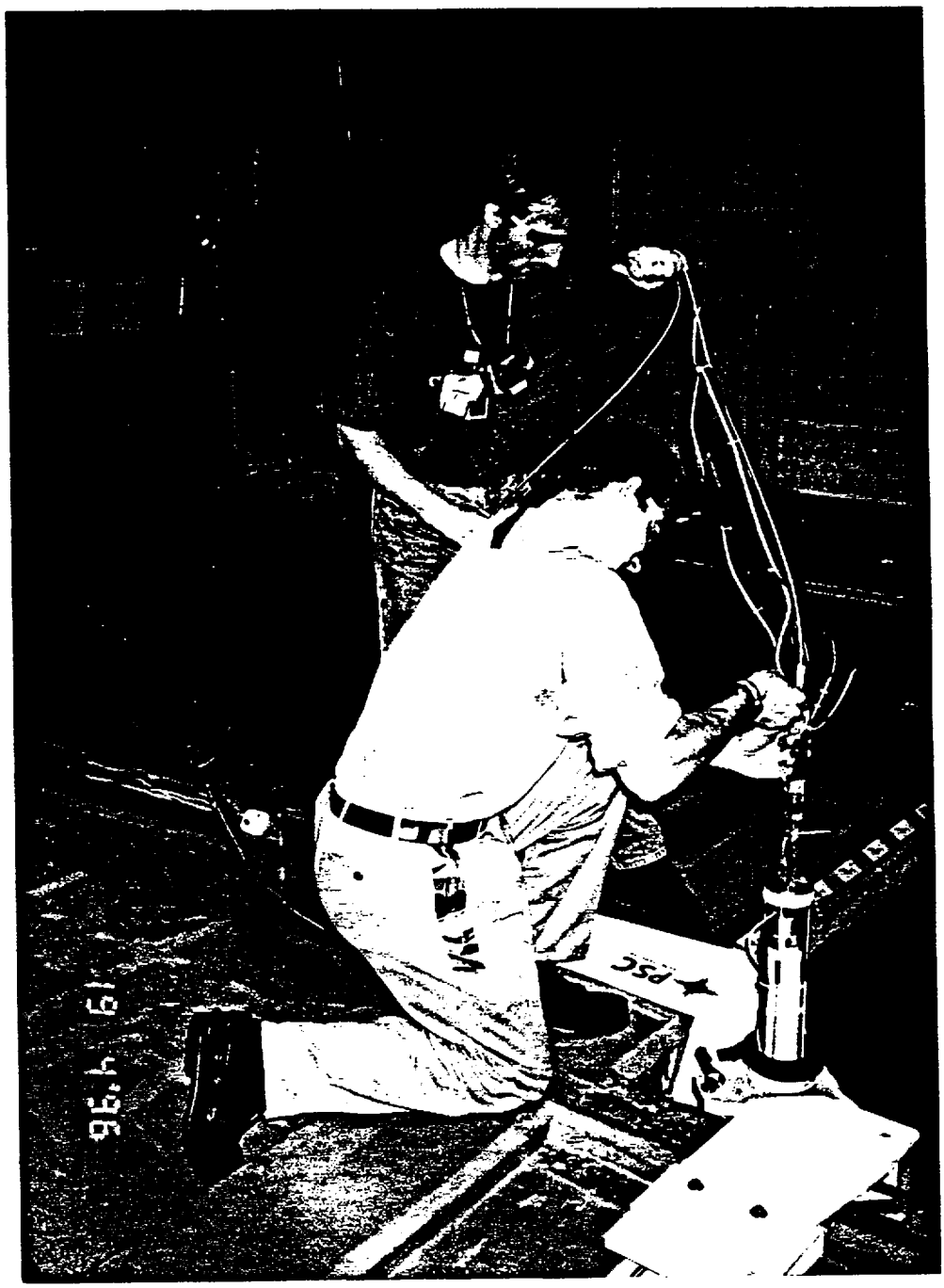


Figure B.1-2 Loading of gamma detector into re-entrant tube.

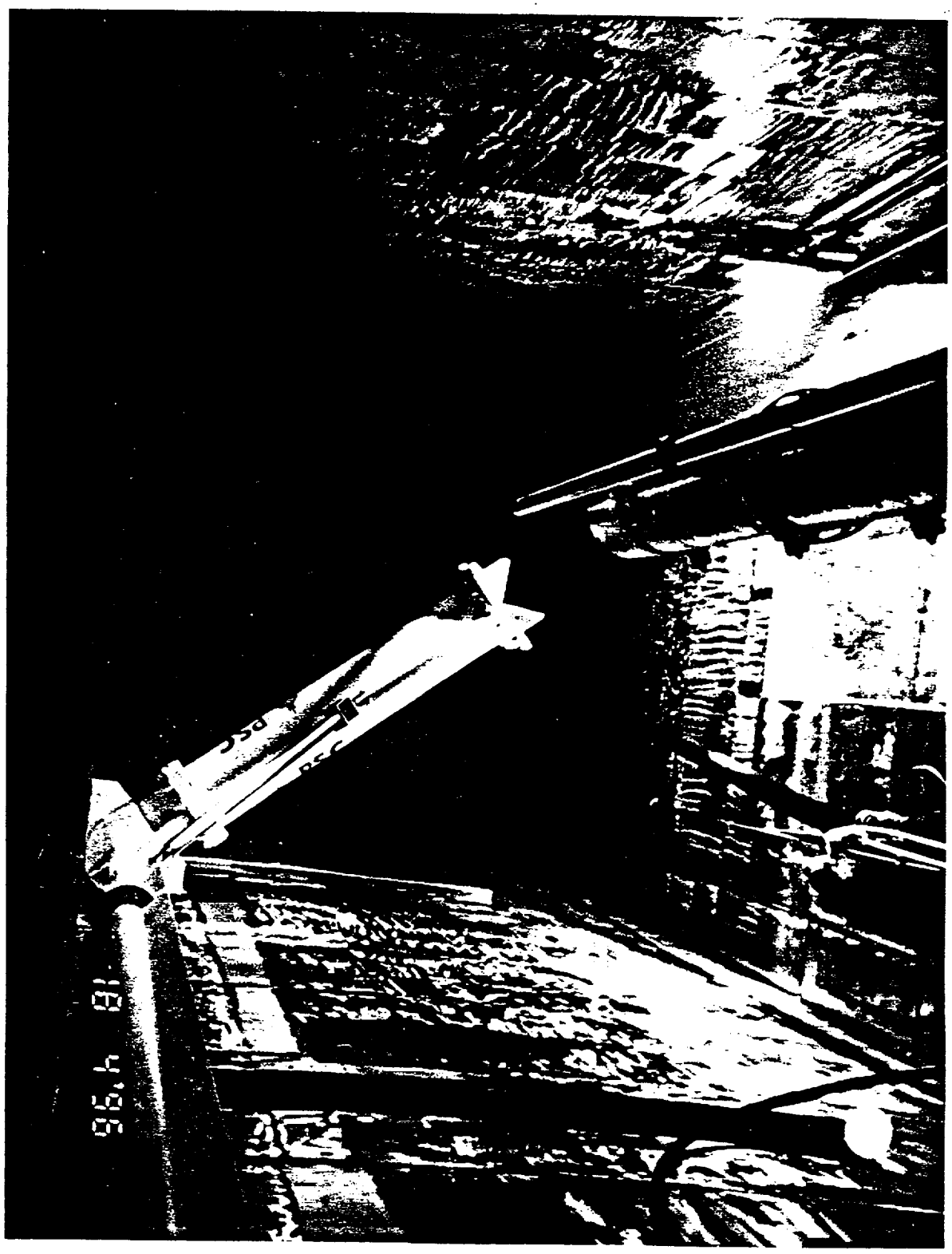


Figure B.1-3 Plan view of gamma collimator and fuel handling machine.

B.1.3 Theory of Operation

The high resolution gamma spectrometric measurements offered by the FuelMaster™ gives the system the ability to resolve and quantify a variety of gamma rays emitted by fission and activation products present in the spent fuel. Radio-isotopes that may be measured by the detection of their gamma emissions include Cs-137, Eu-154, Cs-134, Ce-144 and Ru-106. Individually or in combination, the quantities of these isotopes, built up during irradiation, can be correlated with burnup. As such they are often referred to as "burnup indicators". However, for spent fuel with cooling times in the range 5 to 40 years of which 20 years is typical for US fuels, some of the shorter lived isotopes (Ru-106 and Ce-144) may have decayed away. This leaves Cs-137 as the primary burnup indicator via its β decay daughter Ba_m-137, 661.66* keV gamma ray.

Cs-137 has a half life of 30 years and is a direct fission product with an almost identical fission yield from both uranium and plutonium. It has a linear relationship with burnup when corrected for cooling time and is insensitive to variations in U-235 enrichment, reactor power rating and dwell times. In addition, its long half life means it can easily be measured in fuel cooled to more than 100 years. These characteristics make Cs-137 a particularly good indicator of burnup.

The linear relationship can be expressed in the standard way;

$$\text{Cs-137} = a + b.BU$$

where Cs-137 is the count rate of the 661.66 keV gamma ray corrected to zero cooling time, and "a" and "b" are constants in the linear correlation with burnup BU.

The cooling time, required to correct for the decay of the Cs-137 content in the time between reactor shutdown or discharge from the reactor and measurement, also can be measured by the spectroscopy system using isotopic activity ratios. This additional capability to measure cooling time can be used as a check on the reactor records cooling time and thereby give further confidence in the assembly's irradiation history data.

An example of a typical gamma spectrum, produced on a multichannel analyser (MCA), is shown in figure B.1-4. The log ordinate, counts per channel, is shown against the channel numbers calibrated in energy (keV). The dominance of the 661.66 keV photopeak is shown clearly even on the log scale.

Other algorithms based on HRGS measurements may be brought into use as additional measures of burnup but these are seen, for the reasons given below, as of only secondary importance compared to the Cs-137 technique. The alternative techniques include the use of the activity ratios Cs-134/Cs-137 and Ru-106 x Cs-137/(Cs-134)^2. These techniques, for shorter cooled fuels where all the required isotopes are measurable, offer the advantage that they do no not rely on an absolute measurement as is required for Cs-137. Consequently errors arising from variations in detection efficiency due to changes in detector or electronics performance or in the precise

* Reference: Table of Isotopes, Richard B Firestone (Virginia S Shirley Editor) Eighth Edition, John Wiley & Sons Inc. 1996.

positioning of the fuel relative to the detector should be of less importance. The disadvantages are, however, for the first ratio Cs-134/Cs-137 (a) the ratio has a 2.2 year half life and needs a significant cooling time correction, (b) its correlation with burnup is influenced by the initial U-235 wt.% enrichment and by the reactor power rating, and (c) its application is limited to fuel with cooling times of about 20 years or less due to the decay and disappearance of the shorter lived component, Cs-134.

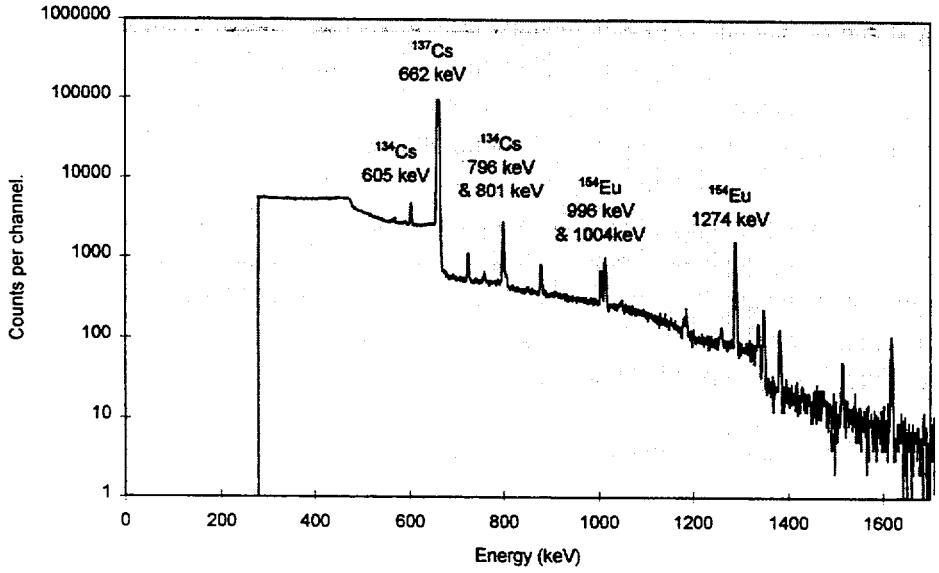


Figure B.1-4 Example of a typical MCA gamma spectrum.

The second activity ratio, Ru-106 x Cs-137/(Cs-134)^2, has the advantage of being an activity ratio method, (i.e. it is insensitive to geometry) but unlike the Cs-134/Cs-137 ratio, it is virtually independent of enrichment and rating and is therefore subject to lower systematic errors. The half life of the ratio is 22 years, giving it a relatively low sensitivity to cooling time correction errors. However, due to the decay of the short half life component Ru-106, this ratio technique can be used only on fuel that has a cooling time of less than about 8 or 9 years. Its general application to a large proportion of US spent fuel is therefore not possible.

B.1.4 Calibration

B.1.4.1 Dependent Calibration. A dependent method of calibration is proposed for burnup measurement systems in the Actinide Only Burnup Credit Topical Report (AOBCTR), Revision 1, March 30, 1997. This approach to calibration has been adopted traditionally for monitoring systems that are used to confirm the consistency of a data set, i.e. to indicate the presence of any outliers. This is considered appropriate for burnup reactor records in which the general accuracy and precision of the data set under test is assumed and that any isolated errors due, for instance, to paper errors would be apparent. These could then be corrected or eliminated from the calibration set. Similarly errors of this type could be corrected or removed from any subsequent measurement set. No account is taken for the possible occurrence of systematic errors or biases in the calibration data sets. The calibrations are, therefore, dependent on the quality of the burnup

records of those spent fuel assemblies selected for the calibration measurement campaigns. Benefits of the dependent approach are that the calibration assemblies are of the same geometry as the fuel to be measured, consequently some systematic effects in the measurement system are accounted for automatically. As mentioned above, it is possible that with an HRGS measurement system other fuel parameters, such as cooling time, can be determined independently to provide supporting validation of the reactor records.

The test proposed in the AOBCTR to qualify the measurement system and to some degree the reactor records uses the following test expression for expected burnup uncertainty, CBU*;

$$CBU = t_{\alpha/2,n-2}\sqrt{\left(\frac{n+1}{n}+\frac{(x-\bar{x})^2}{S_{xx}}\right)\frac{SS_R}{n-2}}$$

where;

$S_{xx} \equiv \sum_{i=1}^{n}(x_i-\bar{x})^2 =$ the sum of differences squared in reactor records burnup about the mean reactor records burnup for the sample population and,

$SS_R \equiv \sum_{i=1}^{n}(y_i-y_{fit})^2 =$ the sum of differences squared in measured burnup from the linear regression value for the sample population.

With the rejection criterion imposed that the CBU is to be less than 10% to a 95% confidence limit and the reactor records uncertainty is assumed to be 5% for 2σ, the measurement system must also offer a measurement uncertainty of better than 5% for 2σ. An assessment has been undertaken to test the results of a recent measurement campaign of 40 assemblies against the dependent system rejection criterion. This is presented in the Measurement Data Section.

B.1.4.2 Independent Calibration. There is interest in using methods of calibrating monitoring systems which are independent of reactor records data. An independent approach, that can be implemented by the FuelMaster™ HRGS system, is to determine the correlation between burnup indicators and burnup by the use of computer burnup inventory codes such as ORIGEN and FISPIN[2]. These codes, established for many years and validated by comparison with experimental destructive analysis data[3, 4], provide inventories of fission products and transuranic nuclides as a function of irradiation history.

Examples of the correlation between two key burnup indicators from both ORIGEN and FISPIN are given in figures B.1-5 and B.1-6. The data is for generic PWR fuel of 5 years cooling time and initial enrichments of 2, 2.5, 3, and 3.5 U-235 wt.%.

Figure B-1.5 shows that there is good agreement between the two codes for the content of the primary burnup indicator Cs-137. Its magnitude has been shown to be consistently predicted by

* Note that *t* distribution tables are required to evaluate $t_{\alpha/2,n-2}$ at the 2 sigma level of significance (i.e. a two tailed significance test).

the different codes and has been validated satisfactorily by destructive analysis to an uncertainty to 1σ of less than 1%,. If the measurement geometry and detection efficiency are well known and are reproducible, Cs-137 can be used to provide a calibration fully independent of operator irradiation history data. In addition, the cooling time needed to correct for the decay of Cs-137 can also be measured independently using gamma spectrometry. The only data required, therefore, for the calibration are the structural details of the fuel assembly which are available to a high quality from the fuel manufacturer.

It is crucial in this approach, however, that no changes occur between the calibration conditions and the measurement conditions. Such changes could include variations in: (i) the detection efficiency, (ii) the measurement geometry, and (iii) the fuel assembly geometry. A measurement procedure that uses this approach should, therefore, include suitable checks to eliminate systematic errors from these possible variations.

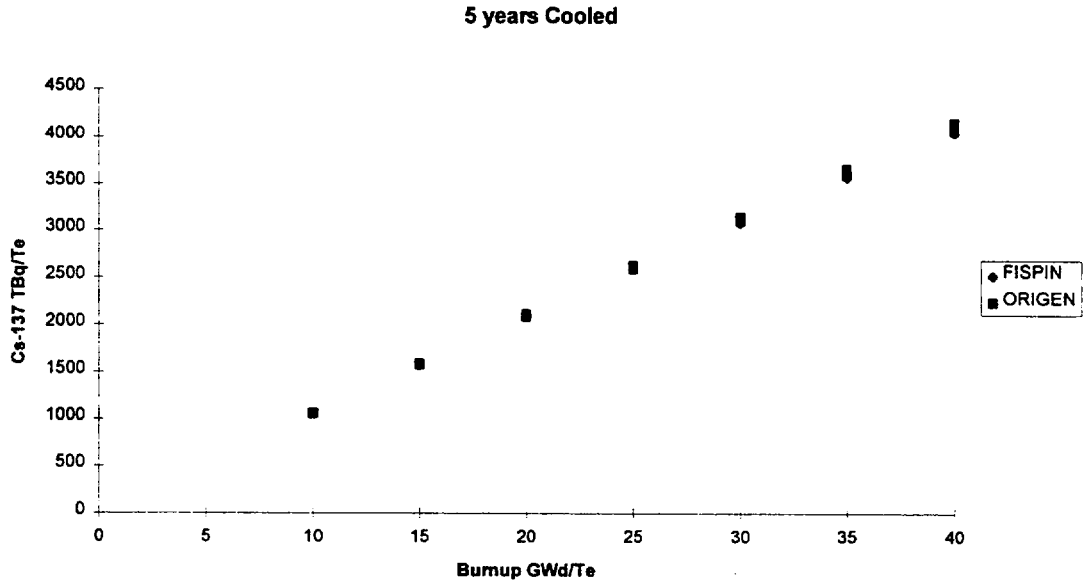


Figure B.1-5 Inventory code derived Cs-137 content per unit uranium mass as a function of burnup.

Figure B.1-6 shows an example of an alternative burnup indicator, Cm-244. This is the primary burnup indicator used by neutron measurement systems via its spontaneous neutron emission. As shown, there is a clear disagreement between the two codes; also, the correlation between Cm-244 and burnup is not linear and there is a strong dependency on the U-235 wt.% initial enrichment. For this reason and because neutron based measurement systems are sensitive to the presence of neutron poisons in the measurement pool, an independent calibration using Cm-244 would be difficult to implement.

In order to achieve an independent calibration based on the preferred burnup indicator, Cs-137, a calibration procedure that is more involved than for a dependent method is still a basic

requirement however. For example, an accurate knowledge and understanding is required of: (i) the fuel assembly parameters, (ii) the gamma attenuation between the source of the gamma emission in the fuel, (iii) the collimator design, (iv) the detector intrinsic efficiency, and (v) the radiometric electronics performance characteristics.

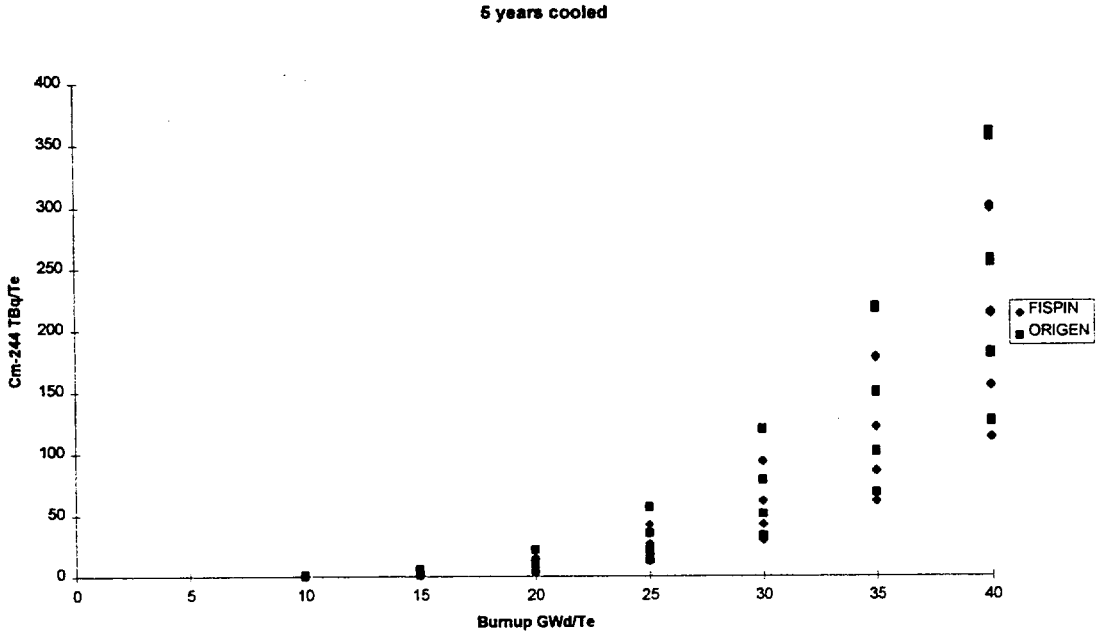


Figure B.1-6 Inventory code derived Cm-244 content per unit uranium mass as a function of burnup.

Considered in the fuel assembly parameters are the Cs-137 activity as a function of burnup at zero cooling time, the activity per unit volume of the fuel assembly, the branching ratio of the Cs-137 661.66 keV gamma ray, and the cooling time of the fuel at measurement. The influence of the fuel structure and the collimator design on the attenuation of gamma rays from the fuel also need to be taken into account. Finally, assessment of the combined effects of the above and the detector and radiometrics electronics performance are necessary to correctly link the measured gamma ray count rate, the quantity of burnup indicator Cs-137 in the fuel to give a measure of burnup. A mixture of analytical calculations and Monte Carlo modelling techniques are likely to be required to satisfactorily carry out this process.

In practice, these processes would be applied by calibrating the collimator and associated counting chain using a National Laboratory traceable gamma sealed source, while both the quantity of Cs-137 and its gamma escape probability in spent fuel would be modelled.

B.1.6 Measurement Procedure

The fuel measurements during 1996 using the demonstration version of the FuelMaster™ were made in a cask loading pit adjacent to a fuel storage pool.

The daily measurement procedure included; (i) detector operational tests using a standardization source prior to the day's fuel measurements, (ii) background radiation measurements, and (iii) axial burnup profile and point gamma spectrometry measurements at several positions along the length of each selected assembly.

In order to minimize systematic errors due to radial burnup profile or radial tilt, the measurement procedure includes turning each fuel assembly through 180 degrees and taking measurements along the assembly on the opposite corner. This allows the measurement of all four faces of the assembly and calculation of the "correct horizontal average". After completion of the measurements, each assembly was returned its original rack location.

Measurements from each position were subsequently combined to determine assembly average values of burnup and cooling time. The standard assembly measurement time was approximately 30 minutes.

As the burnup determination is based on the absolute measurement of the Cs-137 661.66 keV gamma ray count rate, it is crucial that the procedures and geometrical arrangement ensure reproducible positioning of the fuel assembly and consistent detector and associated electronics performance throughout a measurement campaign. To achieve this, the measurement procedure includes suitable checks to eliminate the possibility of systematic errors from these detection efficiency related parameters.

Two methods used to monitor for these effects were (i) confirmation of the detection efficiency of the radiometric system by the measurement of the gamma standardisation source, and (ii) daily measurement of a local reference assembly chosen from additional assemblies available in the same pool. This latter method is very important as it would reveal any changes in the detector/electronics performance, detector positioning in the re-entrant tube, and very importantly fuel positioning in the field of view of the gamma collimator.

B.1.7 Measurement Data

Published results from the US utility PWR measurements using the demonstration FuelMaster™ system include a set of 55 demonstration measurements made in April and a campaign of 40 assemblies carried out under contract in August of 1996. The measurement results presented below are from this first contracted measurement campaign. The results of other commercial measurement made to date have not yet been published. The reactor records data for the first campaign assemblies is given in Table B.1-1.

In accordance with the procedures given above, a reference assembly was chosen and measured each day to confirm the stability of the measurement system and reproducibility of detector and fuel positioning. Table B.1-2 shows the results of these repeat Cs-137 measurements taken over the duration of the measurement campaign.

Table B.1-1 Measured Assemblies - Operator reactor records data.

Fuel Ident. Number	Discharge Date	Meas. Date	Cooling Time Days	Burnup MWd/Te	Initial Enrich. U-235 wt.%
NJ1A02	27-Jan-77	10-Aug-96	7135	17010	2.050
NJ1A03	27-Jan-77	14-Aug-96	7139	17930	2.053
NJ1A05	27-Jan-77	09-Aug-96	7134	16930	2.053
NJ1A06	27-Jan-77	11.Aug-96	7136	17640	2.055
NJ1A07	27-Jan-77	09-Aug-96	7134	16850	2.052
NJ1A08	27-Jan-77	11.Aug-96	7136	17650	2.056
NJ1A09	27-Jan-77	12-Aug-96	7137	17920	2.054
NJ1A10	27-Jan-77	11.Aug-96	7136	17050	2.056
NJ1A11	27-Jan-77	10-Aug-96	7135	17040	2.055
NJ1A12	27-Jan-77	09-Aug-96	7134	16850	2.050
NJ1A13	27-Jan-77	11.Aug-96	7136	17160	2.053
NJ1A14	27-Jan-77	09-Aug-96	7134	16850	2.055
NJ1A15	27-Jan-77	08-Aug-96	7133	16400	2.057
NJ1A16	27-Jan-77	11.Aug-96	7136	17650	2.056
NJ1A17	27-Jan-77	14-Aug-96	7139	17980	2.057
NJ1A18	27-Jan-77	10-Aug-96	7135	16980	2.059
NJ1A19	27-Jan-77	11.Aug-96	7136	17740	2.057
NJ1A21	27-Jan-77	10-Aug-96	7135	17020	2.055
NJ1A22	27-Jan-77	14-Aug-96	7139	17930	2.057
NJ1A24	27-Jan-77	10-Aug-96	7135	16980	2.053
NJ1A25	27-Jan-77	10-Aug-96	7135	17010	2.054
NJ1A27	27-Jan-77	09-Aug-96	7134	16940	2.056
NJ1A42	27-Jan-77	12-Aug-96	7137	17920	2.050
NJ1A43	27-Jan-77	14-Aug-96	7139	17920	2.060
NJ1A44	27-Jan-77	09-Aug-96	7134	16850	2.052
NJ1A45	27-Jan-77	11.Aug-96	7136	17640	2.060
NJ1A50	27-Jan-77	11.Aug-96	7136	17060	2.059
NJ1A52	27-Jan-77	10-Aug-96	7135	17040	2.057
NJ1A55	27-Jan-77	11.Aug-96	7136	17400	2.055
NJ1B03	02-Feb-78	14-Aug-96	6768	28315	2.724
NJ1B18	02-Feb-78	13-Aug-96	6767	28336	2.724
NJ1B19	02-Feb-78	14-Aug-96	6768	28322	2.725
NJ1B28	02-Feb-78	13-Aug-96	6767	28129	2.730
NJ1B36	02-Feb-78	14-Aug-96	6768	28301	2.727
NJ1B47	02-Feb-78	14-Aug-96	6768	28302	2.727
NJ1A41	02-Jan-81	14-Aug-96	5703	23885	2.061
NJ00WT	08-Nov-82	13-Aug-96	5027	31876	3.011
NJ00WU	08-Nov-82	13-Aug-96	5027	31517	3.011
NJ00X6	08-Nov-82	13-Aug-96	5027	31495	3.010
NJ00X8	08-Nov-82	13-Aug-96	5027	31328	3.010

The small variations of the measured count rates, of less than 1% for each point compared to the mean, arise from (i) the statistical error associated with the measured Cs-137, 661.66 keV, count rate, (ii) the combined detector performance and electronics drift, and (iii) variations in the positioning of the detector in the re-entrant tube and in the fuel assembly with respect to the gamma collimator. This ability to reproducibly measure the burnup indicator, Cs-137, within a small error of less than 1%, demonstrates the quality that the system configuration and the measurement procedures introduce.

Table B.1-2 Cs-137, 661.66 keV count rate measurements from reference assembly NJ00WS.

Measurement Date and Time	Cs-137 661.66 keV photopeak count rate (cps)	Variation from mean (%)
09/08/96 16:03	5220 ± 9	+0.1
10/08/96 07:25	5221 ± 8	+0.1
11/08/96 07:34	5243 ± 9	+0.5
12/08/96 18:27	5200 ± 8	-0.3
12/08/96 19:20	5198 ± 8	-0.3
13/08/96 07:32	5247 ± 11	+0.6
14/08/96 17:43	5179 ± 13	-0.7

As all the fuel assemblies being measured in the campaign were to be returned to their original storage racks, it was possible to use all 40 both to test the correlation between the measured and reactor records burnup and as a calibration set. The assembly average Cs-137, 661.66 keV count rate was plotted against the reactor records burnup to establish a correlation curve. From this curve an empirical calibration was determined to give measured burnup as a function of the reactor burnups. A plot based on this calibration giving measured against reactor burnup is shown in figure B.1-7. Note: (i) a single linear curve has been fitted to all the data and (ii) the scatter of the data points about the fitted curve are from the combination of both reactor records and measurements errors.

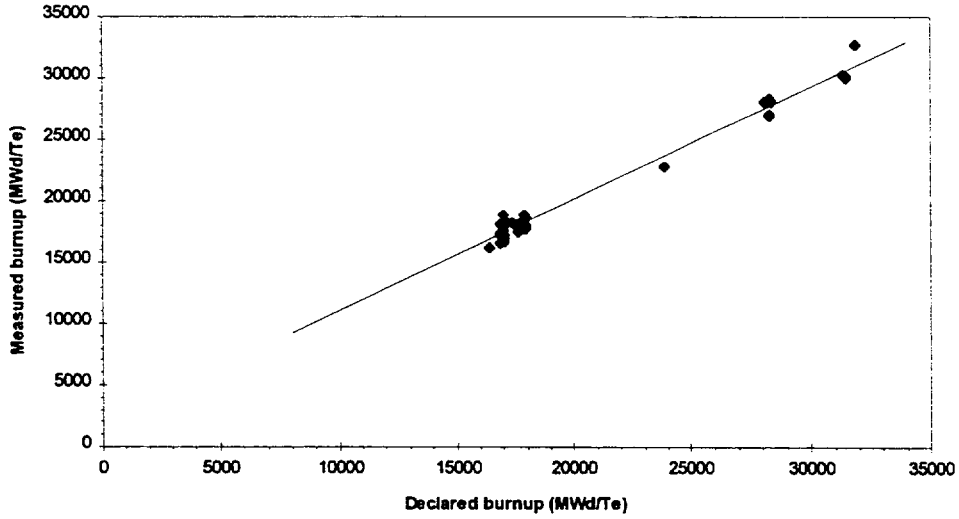


Figure B.1-7 Measured versus reactor records declared burnup for a campaign of 40 assemblies.

The expected burnup uncertainty, CBU, from this dependent calibration system has been calculated using the two tailed *t* test functions as described in the Calibration Section. The CBUs for the 40 assemblies were found to pass the proposed rejection criterion, i.e. the expected burnup uncertainty was found to be less than 10% of the reactor records assembly burnup in all cases. The individual measured burnup and CBU values are presented in Table B.1-3.

Table B.1-3 Measured burnup and CBU rejection criteria test results.

Reactor Records Burnup (MWd/Te)	Measured Burnup (MWd/Te)	CBU	% CBU
31876	32740	1557	4.88
31517	30004	1517	4.81
31495	30124	1519	4.82
31328	30282	1521	4.85
17010	18886	1458	8.57
17930	18542	1459	8.14
16930	17982	1461	8.63
17640	18111	1461	8.28
16850	17225	1464	8.69
17650	17445	1463	8.29
17920	17683	1462	8.16
17050	17206	1465	8.59
17040	16744	1467	8.61
16850	17248	1464	8.69
17160	18238	1460	8.51
16850	16551	1468	8.71
16400	16169	1470	8.97
17650	17871	1462	8.28
17980	17736	1462	8.13
16980	17615	1463	8.61
17740	18247	1460	8.23
17020	18180	1460	8.58
17930	17887	1461	8.15
16980	18035	1461	8.60
17010	18007	1461	8.59
16940	17946	1461	8.63
23885	22845	1459	6.11
17920	18298	1460	8.15
17920	18839	1458	8.14
16850	18105	1461	8.67
17640	17662	1462	8.29
17060	16911	1466	8.59
17040	16652	1467	8.61
17400	18194	1460	8.39
28315	26958	1484	5.24
28336	28056	1495	5.27
28322	28175	1496	5.28
28129	28089	1495	5.31
28301	28361	1498	5.29
28302	27084	1485	5.25

The results show that the performance of the burnup measurement system successfully meets the dependent rejection criterion for this sample population. On this basis the measurement system qualifies for use and would provide confirmation of the reactor record burnups to allow their use with the appropriate burnup curves for cask loading.

B.1.8 Quality Assurance

BNFL Instruments has third party accreditation to ISO 9001 which imposes a thorough control of all company processes. In order to demonstrate consistency with the requirements of the relevant CFRs (10-CFR-50, 10-CFR-71 and 10-CFR-72), a compliance index is being produced for the production FuelMaster™ measurement services for 1997. The index will identify areas where any adaptation of current procedures and instructions relevant to the FuelMaster™ service may be required to be fully compliant with the CFRs.

The Quality Assurance Program will address a number of organisational and procedural issues including;

The company organisation
Design control
Document control
Instructions, procedures and drawings
Control of purchased material, equipment and services
Identification and control of materials, parts and components
Control of special processes
Inspection
Test control
Control of measuring and test equipment
Handling, storage and shipping
Inspection, test and operating status
Nonconforming materials, parts or components
Corrective action
Quality assurance records
Audits.

B.1.9 Summary and Conclusions

The BNFL Instruments FuelMaster™ has demonstrated the practicality of making measurements on spent fuel within a utility's spent fuel pool. Furthermore, this work was carried out with equipment that required a simple equipment installation procedure and with minimal disruption to utility operations.

The results of the measurements have been shown to be compliant with the specified measurement accuracy and rejection criteria proposed in the USDOE OCRWM Topical Report on Actinide Only Burnup Credit.

It can be concluded, therefore, that the FuelMaster™ measurement system is well suited for the verification of reactor records assembly burnup data.

Further information on spent fuel monitoring is available in references 6-10 and on the Web http:\\www.bnflinsts.co.uk\

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Appendix B.2: EPRI's Fork+

Application of the Fork+ radiation measurement system to the verification of burnup records.

B.2.1 Summary

The Fork+ system provides a definitive verification of reactor records for burnup by measuring the residual radiation (gamma-rays and neutrons) from spent fuel assemblies. The Fork+ system incorporates gamma-ray spectroscopy capabilities to provide a determination of assembly burnup that is independent of the reactor records for burnup and initial enrichment. The Fork+ system is an extension of the Fork system, designed at Los Alamos National Laboratory for use by the International Atomic Energy Agency. The Fork+ system retains the overall design of the Fork system to take advantage of the demonstrated operational simplicity and sensitivity of the Fork system. Applications of the Fork system to verification of burnup records at U.S. commercial reactors are described in References 1-3. This appendix is intended to supplement the information in the references, and to describe the additional capability of the Fork+ system.

Gamma-ray spectroscopy permits the identification of a fission product, cesium-137, by its characteristic gamma-ray. Measurement of a fission product can be directly related to the burnup without reference to the reactor record. The Fork+ also has the capability to rapidly measure the gamma-ray yield along the length of an assembly to determine a burnup profile that is used to reduce the uncertainty in the independent measurement of burnup. Analysis of the neutron yield is used to identify with high sensitivity any disagreements between the neutron measurement and the reactor record for burnup of individual assemblies. The Fork+ system fulfills the measurement system design requirements of the Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages (DOE/RW-0472 Rev. 1). The Fork+ system is in the final stages of development by Sandia National Laboratories with the support of the Electric Power Research Institute.

B.2.2 Theory

While undergoing irradiation in the reactor, the fuel assemblies become highly radioactive due to the formation of fission products and neutron capture reactions. After removal from the reactor, the radiation emitted from the assembly decays with the characteristic half-lives of the many radioactive isotopes. The spent fuel assemblies that will be analyzed using the Fork+ have been out of the reactor for over 5 years, which simplifies the analysis of the radiation. After five years of cooling time, cesium-137 is the major gamma-ray emitter. Cesium-137 is produced as a fission product in about six percent of the fissions that occur in the assembly. The burnup (heat output) of the assembly is directly determined by the number of fissions that occurred in the assembly, which is directly related to the cesium-137 content of the assembly at the time of discharge. After five years of cooling time, the only significant neutron emitter is curium-244, which is formed during irradiation by successive neutron capture beginning with uranium-238, and produces

neutrons by spontaneous fission. For cooling times less than five years, curium- 242 is a significant neutron emitter. The production of curium-244 is found to increase with about the fourth power of the burnup. The neutron emission is therefore very sensitive to variations in burnup.

B.2.3 Measurement of Cesium-137

Cesium-137 decays with a half-life of 30 years, resulting in the emission of a gamma-ray of energy 662 keV in 85 percent of the decays. The 662 keV gamma-ray is identified by analyzing the energy spectrum of the gamma-rays emitted from a spent fuel assembly. Standard spectroscopy techniques are employed to produce the gamma-ray energy spectrum, and identify the 662 keV gamma-ray. The concentration of cesium-137 in the spent fuel assembly is determined from the observed intensity by calibration of the spectroscopy system with standard sources of cesium-137 and by calculation of geometric constraints and gamma-ray scattering. The total amount of cesium-137 in the assembly is determined by measuring the relative gamma-ray intensity along the length of the assembly and by integrating the observed concentration. The relative gamma-ray intensity is measured using ion chambers that are collimated to accept gamma-rays from only a small section of the assembly. The cooling time record for the assembly is then employed to extrapolate the total assembly content of cesium-137 back to the content at the time of discharge. The cooling time record is used because there is no direct radiation measurement available to independently determine an accurate cooling time that is effective over the entire cooling interval of interest in spent fuel verification (1 to 100 years).

B.2.4 Independent Calculation of Burnup

The burnup for the assembly can be determined from the total cesium-137 content of the assembly at the time of discharge and its irradiation history. The total number of fissions that have occurred in the assembly is calculated from the known ratio of fissions to cesium-137 (about 16) and the energy released per fission (about 200MeV). The result is the total amount of energy released from the assembly by fission. That result divided by the total amount of uranium in the assembly yields the burnup in the usual units of energy produced per mass of uranium. The irradiation history of the assembly is used to correct for the decay of cesium-137 while in the reactor and for any time out of the reactor between irradiation cycles.

B.2.5 Neutron Yield

The relative neutron yield is measured to provide a sensitive and rapid detection of assemblies for which the neutron yield does not agree with the reactor record for burnup (anomalous assemblies). The sensitivity of this measurement is due to the strong dependence of the neutron yield on the burnup (about the fourth power). The cesium-137 gamma-ray intensity increases as the first power of the burnup, and therefore is not as sensitive as neutrons to variations in burnup.

After the assembly has been out of the reactor for 5 years, the only significant neutron emitter is curium-244, which is formed in the reactor by successive neutron capture beginning with uranium-238. Curium-244 emits neutrons by spontaneous fission with a half-life of 18 years. The production of curium-244 increases with about the fourth power of the burnup. For cooling times of less than five years, the isotope curium-242, with a half-life of 0.45 years, also contributes to the neutron yield. The observed neutron yield can be adjusted for the curium-242 neutrons using well-qualified isotope ratio codes. The yield of neutrons is correlated with burnup by calculated correction factors that make use of the reactor records for burnup, cooling time, and initial enrichment. When the corrected neutron yield is fit to the burnup records with a least-squares power law, the internal variation of burnup is accurately measured, and deviations indicate anomalous assemblies with great sensitivity. The methodology of 6.4.1.1 "Dependent Measurement Systems" can be applied to the neutron yield measurements as a backup to the cesium-137 gamma-ray spectroscopy measurements.

B.2.6 Description of Measurement System

The Fork+ system is similar in design and operation to the Fork system, described in detail in References 1-3. In both systems, Fork and Fork+, gamma-ray and neutron measurements are made simultaneously on two opposite sides of the assembly, with detectors in each of the two arms (or tines of the "fork"). In the Fork+ system, each arm contains a neutron detector in the form of a fission chamber embedded in polyethylene, and a gross gamma-ray detector consisting of an ion chamber. The outputs of the two neutron detectors are added together to average the burnup distribution across the assembly. The two ion chambers are read individually to allow adjustment of the single gamma-ray spectrometer measurement for burnup variations across the assembly. One arm of the detector includes a spectrometer to analyze the energy distribution of the gamma-rays. A cadmium-zinc-tellurium crystal provides the necessary energy resolution to identify the gamma-ray from cesium-137. The gamma-ray sensors are each collimated with tungsten shielding to closely define the field of view on the assembly. Standard commercially available electronic control and readout systems are employed to analyze the detector outputs.

B.2.7 Operation of the Fork+ System

The Fork+ detector is suspended from the fuel handling bridge and immersed in the spent fuel pool at a location just above the fuel rack. The array is moved to the location of the spent fuel assembly to be analyzed. The assembly is raised part way out of the rack until the Fork+ detector is in position for a measurement. The detector is swiveled into contact with assembly and the measurements performed. The assembly is then lowered back into its position in the rack. In general, the neutron and the gamma-ray spectroscopy measurements are performed near the center point of the assembly. A "burnup profile" (relative gamma-ray intensity) is obtained by taking ion chamber measurements at several locations along the length of the assembly.

B.2.8 Proposed Verification Procedure Campaign

The proposed verification procedure would perform neutron and ion chamber measurements at the center level of each assembly. Four assemblies, selected to bracket a range of burnups, are first measured to establish an initial calibration for the neutron system. Gamma-ray spectroscopy and burnup profile measurements would be performed on the four calibration assemblies, and on a sampling basis thereafter to provide the independent measurements of burnup. The sampling could involve a regular pattern, i.e., every ten assemblies, or a selected pattern to satisfy a particular statistical model. Anomalous assemblies identified by the neutron measurements would also be analyzed by gamma-ray spectroscopy and burnup profile measurements.

B.2.9 Fulfilling the Measurement System Design Requirements

The Fork+ system combines both the "dependent" (neutron-based) and "independent" (gamma-ray spectroscopy based) measurement systems. The neutron measurements provide a rapid determination of relative burnup and screen for anomalous assemblies with great sensitivity. The more time consuming independent determinations of burnup would be performed on the calibration and sampling assemblies, assemblies for which the burnup is suspect or crucial (as would be the case for burnup near the loading curve), and anomalous assemblies detected by the neutron measurements. An additional internal calibration is available through the intercomparison of the neutron and gross gamma-ray yield measurements. The neutron measurements are used to check internal consistency, for intercomparison with calibration assemblies, as backup for the "independent" gamma-ray spectroscopy measurements, and to detect anomalous assemblies. The accuracy requirement of 10% in burnup is met by the gamma-ray based measurements by careful control of geometry and by calibration with standard sources.

B.2.10 Conclusions

The Fork+ system uses three measurement techniques: neutron detection, gross gamma-ray detection, and gamma-ray spectroscopy. Neutron detection is used to determine the internal variability of the reactor records for burnup, and to detect anomalous assemblies. Gamma-ray spectroscopy is used for the independent determination of the assembly burnup. The gross gamma-ray detectors are used to determine a relative burnup profile along the length of the assembly, to specify the side-to-side horizontal variation in burnup, and for intercomparison of the assemblies with and without gamma-ray spectroscopy data.

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APPENDIX C

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 D

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.

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