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BURNUP CREDIT IN SPENT FUEL STORAGE AND TRANSPORTATION: REGULATORY PERSPECTIVE ON CURRENT STATUS AND FUTURE DEVELOPMENTS

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ABSTRACT

Allowance in the criticality safety analysis of spent fuel storage and transportation systems for the decrease in fuel reactivity resulting from irradiation is termed burnup credit. Extensive investigations have been performed both within the United States and by other countries in an effort to understand and document the technical issues related to the use of burnup credit. To address technical issues associated with Pressurized Water Reactor (PWR) fuel burnup credit, the U.S. Nuclear Regulatory Commission (NRC) issued Interim Staff Guidance 8, Revision 3, *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks*, in October of 2012. This paper will outline the significant changes in this guidance document from the previous revision, and the technical basis behind them.

Additionally, although burnup credit has been used to some degree in Boiling Water Reactor (BWR) spent fuel pools, it has not yet been requested for BWR spent fuel dry storage or transportation system designs. In anticipation of receiving such requests, the NRC is initiating a long-term BWR burnup credit research project. This project will investigate the various technical issues associated with BWR burnup credit that differ from PWR burnup credit, such as: control blade exposure during irradiation, varying axial moderator density, use of burnable poison rods, varying axial and pin enrichments, and partial length rods. The conclusion of this research will be a set of conclusions and recommendations regarding calculation assumptions and analytical methods for BWR burnup credit, which will be incorporated into revised NRC guidance. This paper will describe the technical issues associated with BWR burnup credit, and outline the plans for the research program to address them.

INTRODUCTION

The United States Nuclear Regulatory Commission (NRC) regulations require that spent nuclear fuel (SNF) remain subcritical during transportation and storage. Historically, NRC has required criticality analyses for transportation and storage systems to assume that the fuel is unirradiated. Unirradiated fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transportation and storage systems. More recently, however, applicants for SNF storage and transportation licenses have designed systems with higher capacity, which require credit for decreased reactivity due to burnup to demonstrate subcriticality. "Burnup credit" refers to accounting for partial or full reduction of SNF reactivity due to the combined effect of the net reduction of fissile nuclides and the production of parasitic neutron absorbing nuclides (non-fissile actinides and fission products) with fuel irradiation.

BACKGROUND

In comparison to the fresh fuel assumption, performing criticality safety analyses for SNF systems that credit burnup require:

1. additional information and assumptions for input to the analysis,
2. additional analyses to obtain the SNF compositions,
3. additional validation efforts for the isotopic depletion and decay software,
4. enhanced validation to address the additional nuclides in the criticality analyses, and
5. verification that the fuel assembly to be loaded meets the minimum burnup requirements made prior to loading the system.

Burnup credit analyses have been limited in the past by the amount of data available to perform validation of the isotopic depletion and criticality safety computer codes used. These data typically consist of 1) destructive radiochemical assays of actual spent fuel samples to validate the isotopic depletion code, and 2) critical experiments representative of spent fuel in a transportation package (i.e., with the actinide and fission product nuclides of interest for burnup credit present in the fuel matrix).

Many radiochemical assay measurements of spent fuel samples have been performed over the past several decades for the purposes of validating reactor core performance and isotopic depletion codes. These samples were assayed for various nuclides important for core performance, primarily the isotopes of uranium and plutonium, as well as several higher actinides. Although there are a large number of measurements available, only a few have values for all of the actinide and fission product nuclides of interest for burnup credit criticality calculations. This makes validation of isotopic depletion codes a challenge, typically resulting in large uncertainties in the calculated composition for a given nuclide.

Historically, critical experiments available for SNF criticality validation have been limited to: 1) fresh low-enriched UO_2 systems, and 2) fresh mixed uranium and plutonium oxide (MOX) systems. These systems are not representative of SNF in a transportation package, as fresh UO_2 systems contain no plutonium, higher actinides, or fission products, and MOX experiments generally do not have plutonium isotopic ratios consistent with that of burned fuel. More applicable experiments have become available recently, as will be discussed later in this paper, but validation of criticality codes for SNF burnup credit calculations remains a significant challenge.

Another challenge associated with SNF burnup credit analyses for transportation packages relates to the assigned burnup loading value associated with each assembly. In the U.S., utilities typically determine actual fuel assembly discharge burnup, based on in-core power measurements, with a low uncertainty (typically on the order of 2% or less). However, there have been multiple instances of fuel assembly misleads, both in spent fuel pools and in dry storage casks. A misloaded assembly will have no effect on maximum system reactivity when the criticality analysis has been performed assuming fresh fuel. However, misloading a fresh or low burnup assembly in a transportation package designed using burnup credit in the criticality analysis can have a significant effect on system k_{eff}, when considering fresh water in-leakage as required by NRC transportation regulations.

NRC's Office of Nuclear Materials Safety and Safeguards (NMSS), which is responsible for the regulation of SNF storage in Independent Spent Fuel Storage Installations (ISFSIs) and in transportation, has had staff guidance available for reviewing pressurized water reactor (PWR) SNF burnup credit applications since 1999. Burnup credit for boiling water reactor (BWR) SNF has not been sought by transportation applicants yet, so NRC has not prepared staff guidance for reviewing such applications.

PWR BURNUP CREDIT GUIDANCE

NRC issued Interim Staff Guidance 8, Revision 2 (ISG-8), *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks*,[1] in 2002. This staff guidance documented the assumptions and methodologies to be used in criticality analyses for SNF storage or transportation systems that rely on burnup credit. Based on the isotopic depletion and criticality code validation available at the time this guidance was published, this guidance recommended taken credit for the major actinides only (^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am). These nuclides represented the bulk of the reduction in k_{eff} due to burnup, and were the best validated in terms of radiochemical assay measurements and critical experiments available. The fission product worth was then retained as a margin to cover uncertainties in the actinide validation, as well as any other uncertainties in the overall burnup credit methodology.

This revision of ISG-8 also recommended a burnup measurement to confirm the calculated burnup value of each assembly. This recommendation was intended to prevent misloads of higher reactivity assemblies into a burnup credit transportation package, which was not previously possible under the fresh fuel assumption. Revision 2 of the ISG states that the measurement can be performed on each assembly, or a representative sample of assemblies, for comparison to the reactor record values.

Since 2002, NRC has conducted a large amount of research in the PWR burnup credit area, primarily through its contractors at Oak Ridge National Laboratory (ORNL). This research focused on isotopic depletion and criticality code validation data and techniques, as well as SNF misload consequences. Part of this research involved obtaining new radiochemical assay and critical experiment data from international research programs related to burnup credit. The database of radiochemical assay data available for validation expanded significantly during this time, particularly for fission products of interest for burnup credit. NRC reviewed and evaluated the available data in a series of reports:

- NUREG/CR-7013, *Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor*,[2]
- NUREG/CR-6968, *Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors*,[3]
- NUREG/CR-6969, *Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation-ARIANE and REBUS Programs (UO2 Fuel)*,[4] and
- NUREG/CR-7012, *Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel*.[5]

During this same time period, NRC also obtained access to a group of critical experiments designed for validating SNF k_{eff} reduction due to major actinides. This actinide criticality validation data is described in detail in NUREG/CR-6979, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*,[6] and is available to applicants from ORNL, subject to execution of a non-disclosure agreement. These experiments were demonstrated to be more appropriate for validating the code-predicted reduction in *keff* due to actinide depletion than fresh UO_2 or other MOX critical experiments.

Additionally, NRC conducted research on consequences of a SNF assembly misload, in terms of Δk_{eff} from the design basis. This research, summarized in NUREG/CR-6955, *Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*,[7] concluded that a low burnup fuel assembly misload can have a significant effect on system k_{eff}. NRC's Office of Nuclear Regulatory Research (RES) also conducted research on the probability of a misload in a transportation package for SNF. This research, summarized in the RES report *Estimating the Probability of Misload in a Spent Fuel Cask*,[8] concluded that transportation package SNF misloads are credible and should be considered in the criticality analysis.

NRC also conducted research regarding burnup credit validation methodologies for both isotopic depletion and criticality codes. This research is summarized in a pair of reports which included recommendations to staff regarding how the results of this research should be incorporated into review guidance.

NUREG/CR-7108, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions*,[9] evaluates several validation techniques using the most complete set of radiochemical assay data available. This report concludes that the Monte Carlo Uncertainty Sampling validation methodology makes best use of the available radiochemical assay data, and calculates reference depletion bias and bias uncertainty values as a function of burnup for a generic cask using SCALE/TRITON with ENDF/B-V and ENDF/B-VII data. Tables 1 and 2 below summarize the reference results. This report also concludes that the Direct Difference method, with appropriate treatment of surrogate data for nuclides not present in all measurements, produces comparable values. NUREG/CR-7108 recommends that applicants for burnup credit transportation packages may use the values in Tables 1 and 2 directly, provided that:

- The applicant uses the same depletion code and cross section library as was used in NUREG/CR-7108 (SCALE/TRITON and the ENDF/B-V or -VII cross section library),

- The applicant can justify that its design is similar to the reference system design used as the basis for the NUREG/CR-7108 isotopic depletion validation, and
- Credit is limited to the 28 specific nuclides listed in Table 3.1 of NUREG/CR-7108.

Table 1: Isotopic *keff* bias uncertainty (Δk_i) for the representative PWR SNF system model using ENDF/B-VII data ($\beta_i = 0$) as a function of assembly average burnup

Burnup Range (GWd/MTU)	**Actinides Only** Δk_i	**Actinides and Fission Products** Δk_i
0-5	0.0145	0.0150
5-10	0.0143	0.0148
10-18	0.0150	0.0157
18-25	0.0150	0.0154
25-30	0.0154	0.0161
30-40	0.0170	0.0163
40-45	0.0192	0.0205
45-50	0.0192	0.0219
50-60	0.0260	0.0300

Table 2: Isotopic keff bias (β_i) and bias uncertainty (Δk_i) for the representative PWR SNF system model using ENDF/B-V data as a function of assembly average burnup

Burnup Range (GWd/MTU)	**β_i for Actinides and Fission Products**	**Δk_i for Actinides and Fission Products**
0-10	0.0001	0.0135
10-25	0.0029	0.0139
25-40	0.0040	0.0165

Additionally, NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (keff) Predictions*,[10] used a combination of techniques to validate burnup credit criticality calculations for a reference transportation package model. This report used a combination of the available low-enriched UO_2, MOX, and HTC critical experiments to determine a criticality bias and bias uncertainty due to major actinides. Since these critical experiments are publicly available to applicants for burnup credit criticality SNF transportation package designs, the report recommends that applicants perform the major actinide portion of the criticality validation using traditional validation techniques on a case-by-case basis.

For validation of criticality codes for k_{eff} reduction due to minor actinides and fission products, NUREG/CR-7109 developed a methodology based on the SCALE *Tools for Sensitivity and Uncertainty Methodology Implementation* (TSUNAMI) code,[11] developed as part of the SCALE code system. This methodology uses the nuclear data uncertainty estimated for each fission product cross section known as the cross section covariance data. These data are provided with the ENDF/B-VII cross section library. The TSUNAMI code is used to propagate the cross section uncertainties represented by the covariance data into k_{eff} uncertainties for each fission product isotope used in a particular application. The theoretical basis of this validation technique is that computational biases are primarily caused by errors in the cross section data, which are quantified and bounded, with a 1σ confidence, by the cross section covariance data. The validity of this theoretical basis is discussed in greater detail in NUREG/CR-7109.

The results demonstrate that, for a generic SNF transportation package evaluated with the SCALE code system, and the ENDF/B-V, -VI, or -VII cross section libraries, the total fission product nuclear data uncertainty (1σ) does not exceed 1.5% of the total minor actinide and fission product worth over the burnup range of interest (i.e., 5 to 60 GWd/MTU). In order to use the 1.5% value directly as a bias, applicants for burnup credit criticality transportation package designs should:

- Use the SCALE code system with the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries,
- Justify that its design is similar to the reference cask design used as the basis for the NUREG/CR-7109 criticality validation, and
- Demonstrate that the credited minor actinide and fission product worth is no greater than 0.1 in k_{eff}.

NUREG/CR-7109 also recommends that, for other well-qualified code systems using ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries, a value of 3.0% of the total minor actinide and fission product worth could be used as a bias. This larger value is considered necessary since validation studies similar to that performed in NUREG/CR-7109 have not been performed for other codes. NRC is conducting further research to determine if the 1.5% value is appropriate for selected code systems other than SCALE.

The recommendations for isotopic depletion and criticality validation from NUREG/CR-7108 and -7109 were incorporated into Interim Staff Guidance 8, Revision 3,[12] published in October of 2012. This revision of the ISG contains additional recommendations related to the assigned burnup loading value associated with SNF assemblies to be loaded into a transportation package. These recommendations are based on the results of the RES misload probability report discussed earlier. The key conclusions of the RES work were that misloads into a transportation package are credible, and that misloads of multiple assemblies in a single package are as likely as single assembly misloads.

The results of the RES work were incorporated into the “Loading Curve and Burnup Verification” section of the ISG. In lieu of performing a direct burnup measurement, applicants may perform misload analyses as part of the package criticality evaluation. These analyses should consider misloading of a single severely underburned assembly, as well as misloading of multiple moderately underburned assemblies. The severely underburned assembly for the single misload analysis should be chosen such that the misloaded assembly reactivity bounds 95% of the discharged PWR fuel population considered unacceptable for loading in a particular storage or transportation system with 95% confidence. The multiple moderately underburned assemblies for this analysis should be assumed to make up at least 50% of the system payload, and should be chosen such that the misloaded assemblies’ reactivity bounds 90% of the total discharged PWR fuel population. Applicants may consider a reduced administrative margin of 2% for the misload analysis.

NRC considers the ISG-8, Revision 3 to be a significant improvement over the previous revision. The recommendations included in this guidance should allow for the transportation of a larger percentage of permanently discharged SNF in the U.S., while maintaining an appropriate degree of conservatism in the analysis, as well as margin of subcriticality.

BWR BURNUP CREDIT

Having completed a major revision to guidance for PWR burnup credit, NRC is beginning a research program on BWR burnup credit. Applicants for BWR SNF transportation package designs have not typically requested burnup credit in the past. BWR SNF storage systems must be able to demonstrate subcriticality in fresh water in the spent fuel pool, as opposed to PWR systems which may rely on soluble boron in the pool water during loading. Also, BWR assemblies have a smaller cross section, leading to more assemblies being stored or transported in a single system. This results in there being more neutron absorber volume (one plate between each pair of assemblies) than for a similarly sized PWR package, generally leading to a lower reactivity than for PWR systems. Therefore, BWR systems typically demonstrate subcriticality conservatively assuming the fuel is unirradiated, and that any integral burnable poison designed into the fuel assembly is not present.

However, SNF transportation applicants have recently indicated a need for some level of burnup credit in BWR packages to counteract increased reactivity from reconfigured fuel under accident conditions, as well as to accommodate higher enrichment fuel or lower areal density neutron absorber plates. In anticipation of receiving burnup credit applications for BWR transportation package designs, NRC has initiated a two-phased approach to investigating BWR burnup credit.

Phase I will consist of evaluating the “peak-reactivity” approach to BWR burnup credit for transportation designs. This approach involves identifying the actual most reactive time in life for a given BWR assembly design, considering the integral burnable absorber composition that results in the highest peak. While less conservative than the fresh fuel/no burnable absorber assumption, this approach is still conservative in that it assumes that all the fuel is at its peak designed reactivity. This approach has been used for some time in spent fuel pool criticality analyses, but has not been applied to transportation systems.

Research conducted in Phase I will initially review and evaluate spent fuel pool BWR burnup credit methodologies, and their applicability to SNF transportation packages. Some issues specific to transportation that are not present in pool analyses are: 1) transportation packages are typically designed universal for all fuel types, while spent fuel pool analyses are only performed for the fuel used at the site, and 2) transportation package designs must consider normal and hypothetical accident conditions which may result in changes to the fuel geometry. This research will consider the extent to which these issues affect the applicability of the peak-reactivity approach to transportation systems. Phase I research will also address technical areas that are critical to the determination of the most reactive time in life of a particular fuel assembly design, including:

- Axial and horizontal burnup profiles appropriate for peak reactivity determination,

- Axial moderator density distribution,
- Control blade history,
- Other fuel design and reactor operating parameters (e.g., fuel temperature, burnable absorber rods, partial length rods),
- Radiochemical assay and critical experiment data available for validation of BWR fuel assembly depletion and criticality codes,
- Methodologies for isotopic depletion and criticality code validation, and
- Interdependence between reactor operating parameters

Phase I research is expected to result in a series of recommendations for updating NRC staff review guidance to include peak reactivity BWR burnup credit.

The second phase of BWR burnup credit research will involve investigating what additional information is needed and which assumptions would change for crediting burnup beyond peak reactivity. Phase II research will potentially involve a larger amount of research, since credit for burnup beyond peak reactivity has not been thoroughly explored for either spent fuel pools or transportation systems. Research conducted during Phase II will include investigations into:

- Differences in appropriate assumptions for fuel design and irradiation parameters (e.g., axial and horizontal profile, axial moderator density, control blade history) between peak reactivity and beyond peak reactivity,
- Increased accuracy of depletion and criticality validation methods appropriate for credit beyond peak reactivity,
- Potential for increased probability and consequences of fuel misloads,
- Methods for burnup verification, and
- Any other changes in analysis methods and initial assumptions required for credit beyond peak reactivity.

Phase II research is expected to result in a series of recommendations on BWR burnup credit beyond peak reactivity, which can be incorporated into revised staff guidance or a Regulatory Guide.

FUTURE DEVELOPMENTS

This paper has discussed a large amount of research and guidance development that has occurred in the past decade, or that will be performed in the near future. When NRC receives the results of the research discussed above, and has an acceptable set of recommendations for revising its guidance, staff will draft a new guidance document. This document will either consist of consolidated staff guidance on SNF burnup credit (e.g., and ISG covering both PWR and BWR burnup credit), or a new Regulatory Guide on burnup credit in transportation.

To the extent practicable, NMSS staff will consult with other NRC Offices responsible for reviewing SNF criticality analyses (Office of Nuclear Reactor Regulation, Office of New Reactors), to maintain consistency within our guidance. There will likely continue to be differences in how burnup credit is applied in these different regulated areas, given the different environments (i.e., within a controlled environment at a reactor site for spent fuel pool analyses versus in the public domain for transportation analyses).

Other future developments in burnup credit analysis will likely include improvements to the computational tools used for burnup credit criticality analyses. Current transportation package burnup credit criticality analyses involve a large amount of analysis using complicated and time-consuming code input methods, which also involve a considerable amount of effort in the review phase. NRC is exploring the possibility of developing new computational tools for burnup credit criticality analyses for transportation package designs, which would make computer modeling less complex. Some of the features that NRC is seeking include:

- Simpler input methods for fuel design information, including irradiation parameters,
- Automated depletion modeling with built-in or custom axial profiles,
- Ability to model assembly-by-assembly parameters, for modeling actual package loading configurations or misloads,
- Ability to easily input existing discharged fuel data into a package model.

NRC is seeking these advancements to make it much easier for applicants to create, and for NRC staff to review and confirm, detailed criticality models for burnup credit criticality analyses of SNF transportation packages.

CONCLUSIONS

NRC has concluded a large amount of research on PWR burnup credit in the past decade, which has now been incorporated into staff review guidance. NRC is beginning a project to conduct similar research for BWR burnup credit in the near future, with the goal of having consolidated PWR and BWR guidance for transportation system burnup credit criticality analyses. Enhancements to computational tools used for burnup credit analysis should more realistically account for the decreased reactivity in the SNF due to burnup and to make criticality analyses for SNF transportation packages much easier for applicants to perform, and for NRC staff to review and confirm.

DISCLAIMER NOTICE

The author is solely responsible for the opinions, recommendations, and conclusions expressed herein. This paper does not necessarily reflect the views of the U.S. Nuclear Regulatory Commission.

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