JANUARY 5, 2015

ATTACHED IS A MEMORANDUM DATED FEBRUARY 6, 2009, FROM WILLIAM RECKLEY, NRO/ARP, TO GREGORY CRANSTON, NRR/DSS/RSB, AND JOSEPH DONOGHUE, NRO/DSRA/NPCB, RE: REVIEW GUIDANCE RECOMMENDATIONS FOR SPENT FUEL POOL CRITICALITY SAFETY ANALYSIS. THE ENCLOSURES TO THE MEMORANDUM ARE ALSO INCLUDED IN THE ATTACHMENT. THE MEMORANDUM AND THE ENCLOSURES (ADAMS PACKAGE ML090290031) WERE ORIGINALLY PROFILED AS NON-PUBLICLY AVAILABLE (SENSITIVE INTERNAL INFORMATION). INTEREST IN THE MEMORANDUM RESULTED IN A REVIEW IN LATE 2014 AND IT WAS DETERMINED THAT THE MEMORANDUM AND ITS ENCLOSURES COULD BE RE-PROFILED AS NON-SENSITIVE, PUBLICLY AVAILABLE. THE MEMORANDUM AND ITS ENCLOSURES WERE REVIEWED BY THE ORIGINAL AUTHOR (WILLIAM RECKLEY) AND IT WAS DETERMINED THAT THE MATERIAL NO LONGER WARRANTED BEING WITHHELD FROM PUBLIC DISCLOSURE. THE ATTACHED INCLUDES THE MATERIAL IN THE ORIGINAL ADAMS PACKAGE, MODIFIED ONLY BY REMOVING THE MARKING ("INTERNAL USE ONLY") FROM ENCLOSURE 2.

William Reckley
1/5/15

February 6, 2009

MEMORANDUM TO: Gregory V. Cranston, Chief
Reactor Systems Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Joseph E. Donoghue, Chief
Reactor Systems, Nuclear Performance and Code Review Branch
Division of Safety Systems and Risk Assessment
Office of New Reactors

FROM: William D. Reckley, Chief ***/RA/***
Projects and Technical Review Branch
Advanced Reactor Program
Office of New Reactors

SUBJECT: REVIEW GUIDANCE RECOMMENDATIONS FOR SPENT FUEL POOL CRITICALITY SAFETY ANALYSIS

This memorandum follows up on a commitment made on October 8, 2008, during an interoffice staff working meeting for the acceptance screening review of a proposed generic issue on review guidance for storage pool criticality safety. In particular, Outcome item d in the meeting minutes of Enclosure 1 notes that Dr. Donald Carlson of my staff had volunteered (with my approval) to issue a finalized discussion paper that presents the issues in terms of recommended enhancements to the review guidance for spent fuel pool criticality safety analysis. That discussion paper is transmitted herewith as Enclosure 2 in fulfillment of that commitment. It bears noting here that a major conclusion from the acceptance review meeting was that the proposed issue did not meet the acceptance criterion that generic issues must be issues that cannot be addressed "through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives." The discussion paper; therefore, seeks to support staff efforts to address the underlying issues through guidance improvements to be developed outside the processes of the Generic Issues Program. The following paragraphs convey highlights from the paper.

The recommended guidance improvements discussed in the paper would be established by consolidating physically relevant insights gained from review experience and from numerous studies performed in recent years for analyzing spent fuel criticality safety in a variety of out-of-core settings, including pools, cask systems, repository systems, and reprocessing plants, and would thus have broad applicability to the review of all such systems. Used to update and supplement the respective programmatic review guidance presently used for pools and other

CONTACT: Donald E. Carlson, NRO/ARP
301-415-0109

spent fuel systems, the consolidated interoffice review guidance would enable cross-cutting gains of consistency, transparency, and effectiveness in the Nuclear Regulatory Commission (NRC) staff's reviews of analyses submitted to address the respective criticality safety requirements in Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR Part 50) (storage pools), 10 CFR Part 63 (repository systems), 10 CFR Part 70 (reprocessing plants), 10 CFR Part 71 (transport casks) and 10 CFR Part 72 (storage casks).

The NRC technical staff responsible for reviewing the criticality safety of spent fuel management systems are housed in five divisions and three offices. Those who review the criticality safety analyses for at-reactor storage pools reside respectively in the Division of Safety Systems of the Office of Nuclear Reactor Regulation (NRR) and in the Division of Safety Systems and Risk Assessment of the Office of New Reactors (NRO). The reviewers of such analyses for spent fuel in transport and storage cask systems, repository systems, and eventual reprocessing plants are housed respectively in the divisions of Spent Fuel Storage and Transport, High Level Waste Repository Safety, and Fuel Cycle Safety and Safeguards of the Office of Nuclear Material Safety and Safeguards (NMSS). Yet another NRC organization, the Division of System Analysis in the Office of Nuclear Regulatory Research (RES), houses the technical staff who support such review functions by providing related research products (e.g., criticality computer codes, experimental data, analytical studies) as requested by any of the aforementioned divisions in NRR, NRO, or NMSS.

The NRC has a strategic safety goal to "ensure adequate protection of public health and safety and the environment." One of the intended strategic outcomes of that goal is the prevention of any inadvertent criticality events. To that end, various NRC regulations and guidance documents have been established over the years to help ensure that fissile materials, including used or spent fuel, remain subcritical in all out-of-core settings. While certain regulatory requirements for nuclear criticality safety also address consequence mitigation measures (e.g., criticality alarms, worker evacuation, shielding), the primary emphasis is on preventing such events by maintaining appropriate subcritical safety margins.

As noted by NRR in a recent letter, many spent fuel storage pool facilities have been converting in recent years to new pool rack configurations in order to maximize storage capacity. These higher capacity pool configurations necessitate the use of more complex methods to analyze criticality safety. The growing complexity has led the NRR staff to further question various aspects of the analyses, such as how modeling assumptions and approximations are justified, how computer codes are validated against applicable experimental data, and how validation-derived code biases and uncertainties are evaluated and applied in determining subcritical safety margins for pools. Two Commission papers have noted the emergence in recent years of similar trends in the cask systems reviewed by NMSS for spent fuel storage and transport, whereby various proposals, such as those involving increased cask capacities, reduced cask loading restrictions, or transport in dry storage canisters that lack absorber plates, tend to rely on the use of fewer modeling conservatisms and greater overall realism and complexity in the analysis of criticality safety.

The above noted trends involving both pools and casks have also drawn growing attention to the fact that the criticality analysis practices approved by reviewers in NRR and NRO for at-reactor fuel storage pools tend to differ substantially from those approved by reviewers within NMSS for casks and other systems that eventually hold the same spent fuel. While certain differences may be warranted by safety and licensing considerations unique to the respective systems and regulatory domains, it is nevertheless clear that all domains share the same fundamental need to calculate effective neutron multiplication factors, k_{eff}, for similar water-moderated configurations of spent fuel. In particular, neutron moderation by water must be evaluated in the licensing basis criticality safety analyses for dry cask storage, transport, and disposal systems in order to address in-pool cask loading, unloading, and/or repair operations as well as the potential for water ingress to occur in hypothetical transport cask accidents or after centuries of disposal canister residence in a geologic repository.

The frequently shared use of criticality analysis codes and validation data further attests to the high degree of neutronic similarity between the fuel configurations analyzed for the respective domains. As discussed in the paper, the analysis practices approved for pools versus casks differ mainly in the levels of rigor and conservatism that characterize the modeling assumptions and approximations, code validation processes, and bias and uncertainty treatments applied to the respective systems. Related differences are also seen in the types and levels of effort typically applied to the respective reviews. Such differences include for example the fact that NMSS reviewers frequently perform confirmatory calculations of the limiting k_{eff} values for casks, whereas such confirmatory calculations are not routinely included in NRR's reviews for pools. In response to interest from the Commission, technical staff in the affected branches of NRR, NRO, NMSS, and RES held a workshop in February 2008 and established an interoffice discussion forum devoted to mutually understanding their respective review and approval practices for pool versus cask criticality safety analyses and to exploring the resolution of differences where appropriate. This paper seeks to support the staff engaged in those efforts by recommending the consolidation of unified review guidance for analyzing the criticality safety of all spent fuel management systems as licensed per the applicable requirements in 10 CFR Parts 50, 63, 70, 71, and 72. The envisioned guidance would also benefit related staff efforts on knowledge management and would be useful in addressing related concerns and criticisms raised by industry groups. Having extensively supported NMSS with computer codes, validation data, and analytical studies on the criticality analysis of spent fuel in casks since the late 1990s, RES and Oak Ridge National Laboratory have well proven capabilities to effectively support the establishment of consolidated interoffice guidance as proposed herein. While the initial focus would be on analysis practices specific to the fuel discharged from today's operating power reactors, the evolving scope of the consolidated review guidance would be extended when needed to address the anticipated emergence of future reactors and fuel cycles, which could eventually include non-light-water reactors and fuel reprocessing plants.

The unified guidance is envisioned as taking the form of a NUREG or NUREG/CR report that compiles and summarizes relevant insights gleaned from review experience and from the many technically related studies that have been performed and published in recent years. The report would be updated as needed to address residual guidance gaps and evolving review issues and insights. The report would then be incorporated by reference into the appropriate review guidance documents (e.g., standard review plans, interim staff guidance, regulatory guides, etc.) used for the respective systems and regulatory domains. Also included in the consolidated guidance report would be a general discussion on the allocation of review effort and analytical rigor commensurate with case-specific needs as evaluated in the context of risk-informed, performance-based regulation. For example, greater care and rigor would generally be warranted in cases where safety margins or compliance margins are smaller and associated material configurations are more probable. Conversely, relatively little review effort would be warranted for cases where simple conservative methods and assumptions suffice to show substantial subcritical safety margins or to show safety even in configurations clearly more reactive than any that would be deemed credible or remotely probable.

Criticality safety analyses are potentially susceptible to various kinds of potentially non-conservative errors and omissions and associated uncertainties in calculating the maximum or limiting k_{eff} values of spent fuel systems under the various conditions considered for regulatory compliance and safety assessment. To illustrate the situation, thirteen analysis areas that would potentially benefit from the development of consolidated review guidance are described in the paper. These analysis areas are listed in summary below along with highly provisional upper and lower estimates of the amounts by which the k_{eff} values calculated for hypothetically identical configurations of spent fuel might vary, between non-conservative (i.e., potentially underpredicting) and highly conservative (i.e., reliably overpredicting), absent clear guidance:

Burnup credit isotopics:	2% – 5%	Bundles with removed pins:	0.5% – 1.5%
Burnup credit validation:	1% – 3%	Cooling time Am-241 credit:	0% – 0.5%
Rodded burnup histories:	1% – 3%	Actual fuel pin conditions:	0.2% – 0.5%
Spatial burnup profiles:	1% – 2%	Monte Carlo undersampling:	0% – 1%
Pin burnup modeling:	0.5% – 1.5%	Wall reflection effects:	0% – 0.5%
Spent fuel record accuracy:	0.3% – 1%	Offsetting conservatisms:	0% – 2%
Absorber plate granularity:	0.3% – 1%		

Summing to 7 to 22 percent, the estimated potential variations in calculated k_{eff} appear easily large enough (e.g., in relation to typical 5% subcritical safety margins) to warrant attention to such areas when considering needs and priorities for establishing associated review guidance. In closing, I would like to recommend that NRO and NRR work with NMSS in considering the discussion paper as part of the basis for a potential user-need memorandum requesting RES assistance in the development of review guidance improvements, as suggested under Outcome item c-2 in the meeting minutes of Enclosure 1. I would further like to commend Dr. Carlson's efforts in communicating his insights and recommendations through the discussion paper. Please feel free to contact Dr. Carlson at 301-415-0109 with any related questions or comments.

Enclosures:
As stated

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Meeting Minutes
Proposed Generic Issues on Review Guidance
for Storage Pool Criticality Safety

Meeting Date: October 8, 2008

1. John Kauffman started the meeting by giving a brief overview of the Generic Issues Program (GIP), and the criteria for issues that belong in the program, and possible outcomes for an Acceptance Review.

2. Donald Carlson provided a brief overview of the proposed issue, which was followed by lengthy discussions of the issue, and possible solutions.

3. Outcomes

    a. There was general agreement that the proposed issue does not meet GIP Criterion #3, in that, the issue can be readily addressed by existing regulatory programs, i.e. by establishing improved technical review guidance and using it within the context of existing regulations and regulatory processes. Moreover, it was agreed that revised and more detailed technical guidance is warranted.

    b. There was a brief discussion of how the proposed issue may also not meet GIP Criterion #1. Specifically, it is not clear that the issue significantly affects public health and safety, the common defense and security, or the environment. It was noted that such determinations are and will continue to be considered and confirmed on a case by case basis through the Office of Nuclear Reactor Regulation's (NRR's) and the Office of New Reactor's (NRO's) existing licensing and regulatory processes (including B5b). Because of this, and the fact that the issue does not meet GIP Criterion #3, it will not be necessary to document a conclusion on the proposed issue with regard to Criterion #1.

    c. There was general consensus that the proposed issue should exit the GIP and could be addressed via the following activities:

        1. review guidance improvements could be implemented through Interim Staff Guidance updates to Standard Review Plan 9.1.1 and its references, through Regulatory Guide (RG) modifications (e.g., RG-1.13), through a new "desk reference" for technical reviewers, a new branch technical position memorandum, and a generic communication. Also discussed in this context were considerations for making the guidance public (or not) and a perceived potential for back fit implications depending on how the guidance improvements are implemented.

        2. to ensure that the issue not only can be, but will, be addressed by establishing and using improved technical review guidance; NRR and NRO, and possibly the Office of Nuclear Material Safety and Safeguards (NMSS), should consider issuing a joint user-need memo asking the Office of Nuclear Regulatory Research (RES) to develop technical references and propose review guidance improvements.

Enclosure 1

3. NRR, NRO, and NMSS continue their recently initiated efforts to better converge their historically divergent respective approaches to spent fuel criticality analysis. This would be facilitated by active staff participation in knowledge management activities organized through an interoffice community of practice for spent fuel criticality safety.

d. Donald Carlson volunteered to issue a retitled Revision 1 of his discussion paper that reflects the issue as one of emphasizing enhancements to the review guidance versus deficiencies and forward this revised version via memorandum to the responsible NRO and NRR managers for consideration in developing revised guidance for reviewers.

Attendees:
Mourad Aissa, RES
Edward Baker, NRO
Andrew Barto, NMSS
Donald Carlson, NRO
John Kauffman, RES
Diane Jackson, NRR
Jim O'Driscoll, NRO
William Reckley, NRO
Christopher Van Wert, NRO
Carl Withee, NMSS
Kent Wood, NRR

Discussion Paper

Recommendations for Review Guidance on Spent Fuel Pool Criticality Safety Analysis

(Revision 1)

January 28, 2009

Dr.-Ing. Donald E. Carlson
Senior Project Manager
Advanced Reactor Program
Office of New Reactors

Key Words: Criticality safety, spent fuel[1], used fuel[1], fuel burnup, burnup credit, fission, reactivity decrement, fuel pin, fuel bundle, poison rod, nuclide, actinide, fission product, neutron multiplication, code validation, subcritical margin, storage pool, storage cask, transport cask, waste repository, reprocessing plant, poison plate, Boral, safety analysis, review guidance, safety evaluation.

Purpose

This discussion paper recommends the establishment of improved review guidance for the criticality safety analysis of spent fuel[1] in storage pools. The recommended guidance improvements will consolidate physically relevant insights gained from review experience and from numerous studies performed in recent years for analyzing spent fuel criticality safety in a variety of out-of-core settings, including pools, cask systems, repository systems, and reprocessing plants, and will thus have broad applicability to the review of all such systems. Used to update and supplement the respective programmatic review guidance presently used for pools and other spent fuel systems, the consolidated interoffice review guidance will enable cross-cutting gains of consistency, transparency, and effectiveness in the Nuclear Regulatory Commission (NRC) staff's reviews of analyses submitted to address the respective criticality safety requirements in Parts 50, 63, 70, 71, and 72 of Title 10 of the *Code of Federal Regulations* (10 CFR).

Introduction

The NRC has a strategic safety goal to "ensure adequate protection of public health and safety and the environment." One of the intended strategic outcomes of that goal is the prevention of any inadvertent criticality events. To that end, various NRC regulations and guidance documents have been established over the years to help ensure that fissile materials, including used or spent fuel, remain subcritical in all out-of-core settings. While certain regulatory requirements for nuclear criticality safety also address measures for the mitigation of event consequence (e.g., criticality alarms, worker evacuation, shielding), the primary emphasis is on preventing such events by maintaining appropriate subcritical safety margins.

[1] Throughout this paper, the terms "spent fuel," "used fuel," and "burned fuel" are used interchangeably in referring to any irradiated fuel that has been discharged from a reactor core, either temporarily or permanently, including fuel that has not reached its intended final level of burnup.

The NRC technical staff responsible for reviewing the criticality safety of spent fuel management systems are housed in five divisions and three offices: Those who review the criticality safety analyses for at-reactor storage pools reside respectively in the Division of Safety Systems of the Office of Nuclear Reactor Regulation (NRR) and in the Division of Safety Systems and Risk Assessment of the Office of New Reactors (NRO). The reviewers of such analyses for spent fuel in transport and storage cask systems, repository systems, and eventual reprocessing plants are housed respectively in the divisions of Spent Fuel Storage and Transport, High Level Waste Repository Safety, and Fuel Cycle Safety and Safeguards of the Office of Nuclear Material Safety and Safeguards (NMSS). Yet another NRC organization, the Division of System Analysis in the Office of Nuclear Regulatory Research (RES), houses the technical staff who support such review functions by providing related research products (e.g., criticality computer codes, experimental data, analytical studies) as requested by any of the aforementioned divisions in NRR, NRO, or NMSS.

As recently noted by NRR [1] in reference to Information Notice 2005-12 [2], many spent fuel storage pool facilities have been converting in recent years to new pool rack configurations in order to maximize storage capacity. These higher capacity pool configurations necessitate the use of more complex methods to analyze criticality safety. The growing complexity has led the NRR staff to further question various aspects of the analyses, such as how modeling assumptions and approximations are justified, how computer codes are validated against applicable experimental data, and how validation-derived code biases and uncertainties are evaluated and applied in determining subcritical safety margins for pools. Two Commission papers [3, 4] have noted the emergence of similar trends in the cask systems used for spent fuel storage and transport, whereby various proposals, such as those involving increased cask capacities, reduced cask loading restrictions, or transport in dry storage canisters that lack absorber plates, rely on the use of fewer modeling conservatisms and greater overall realism and complexity in the analysis of criticality safety.

The above noted trends involving both pools and casks have also drawn growing attention to the fact that the criticality analysis practices approved by reviewers in NRR and NRO for at-reactor fuel storage pools tend to differ substantially from those approved by reviewers within NMSS for casks and other systems that eventually hold the same spent fuel [5, 6]. While certain differences may be warranted by safety and licensing considerations unique to the respective systems and regulatory domains, it is nevertheless clear that all domains share the same fundamental need to calculate effective neutron multiplication factors, k_{eff}, for similar water-moderated configurations of spent fuel. In particular, neutron moderation by water must be evaluated in the licensing basis criticality safety analyses for dry cask storage, transport, and disposal systems in order to address in-pool cask loading, unloading, and/or repair operations as well as the potential for water ingress to occur in hypothetical transport cask accidents or after centuries of disposal canister residence in a geologic repository.

The frequently shared use of criticality analysis codes and validation data further attests to the high degree of neutronic similarity between the fuel configurations analyzed for the respective domains. As further discussed later in this paper, the analysis practices approved for pools versus casks differ mainly in the levels of rigor and conservatism that characterize the modeling assumptions and approximations, code validation processes, and bias and uncertainty treatments applied to the respective systems. Related differences are also seen in the types and levels of effort typically applied to the respective reviews. Such differences include for example the fact that NMSS reviewers frequently perform confirmatory calculations of the limiting k_{eff} values for casks, whereas such confirmatory calculations are not routinely included in NRR's reviews for pools.

In response to interest from the Commission [7, 8, 9], technical staff in the affected branches of NRR, NRO, NMSS, and RES held a workshop in February 2008 and established an interoffice discussion forum devoted to mutually understanding their respective review and approval practices for pool versus cask criticality safety analyses and to exploring the resolution of differences where appropriate. This paper seeks to support the staff engaged in those efforts by recommending the consolidation of unified review guidance for analyzing the criticality safety of all spent fuel management systems as licensed per the applicable requirements in 10 CFR Parts 50 (storage pools), 63 (repository systems), 70 (reprocessing plants), 71 (transport casks), and 72 (storage casks). The envisioned guidance would also benefit related staff efforts on knowledge management and would be useful in addressing related concerns and criticisms raised by industry groups. Having extensively supported NMSS with computer codes, validation data, and analytical studies on the criticality analysis of spent fuel in casks since the late 1990s, RES and Oak Ridge National Laboratory (ORNL) have well proven capabilities to effectively support the establishment of consolidated interoffice guidance as proposed herein. While the initial focus would be on analysis practices specific to the fuel discharged from today's operating power reactors, the evolving scope of the consolidated review guidance would be extended when needed to address the anticipated emergence of future reactors and fuel cycles, which could eventually include non-light-water reactors and fuel reprocessing plants.

The unified guidance is envisioned as taking the form of a NUREG or NUREG/CR report that compiles and summarizes relevant insights gleaned from review experience and from the many technically related studies that have been performed and published in recent years. The report would be updated as needed to address residual guidance gaps and evolving review issues and insights. The report, either in its entirety or by selected portions, would then be incorporated by reference into the appropriate review guidance documents (e.g., standard review plans, interim staff guidance, regulatory guides, etc.) used for the respective systems and regulatory domains. Also included in the consolidated guidance report would be a general discussion on the allocation of review effort and analytical rigor commensurate with case-specific needs as evaluated in the context of risk-informed, performance-based regulation. For example, greater care and rigor would generally be warranted in cases where safety margins or compliance margins are lower and associated material configurations are more probable. Conversely, relatively little review effort should be necessary for cases where simple conservative methods and assumptions suffice to show substantial subcritical safety margins or to show safety even in configurations far more reactive than any that would be deemed credible or remotely probable.

Criticality safety analyses are potentially susceptible to various kinds of potentially non-conservative errors and omissions and associated uncertainties in calculating the k_{eff} values for spent fuel systems under the various conditions considered for regulatory compliance and safety assessment. To illustrate the situation, 13 analysis areas that would potentially benefit from the development of consolidated review guidance are described in this paper. These analysis areas are listed in summary below along with highly provisional upper and lower estimates of the amounts by which the k_{eff} values calculated for hypothetically identical configurations of spent fuel might vary, between non-conservative (i.e., potentially under-predicting) and highly conservative (i.e., reliably over-predicting), absent clear guidance:

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Summing to 7 to 22 percent, the estimated potential variations in calculated k_{eff} appear easily large enough (e.g., in relation to typical 5% subcritical safety margins) to warrant attention to such areas when considering needs and priorities for establishing associated review guidance. Each of these analysis areas is discussed in the following section. The paper then concludes with some preliminary thoughts and observations on potential considerations for applying consolidated review guidance to criticality safety analyses across various regulatory and technical settings.

Potential Guidance Areas for Spent Fuel Criticality Safety Analysis

Thirteen examples of criticality safety analysis areas that would or may benefit from consolidated review guidance are discussed in the following subsections and Appendix A. The examples are not intended to be comprehensive. It is therefore likely that additional areas will eventually be found to merit discussion in this context (e.g., the potential for systematic fuel pool or cask misloadings due to erroneous records and/or procedures, poison plate axial coverage, eccentric fuel positioning in racks, burnup verification). It is hoped that the following discussions will elicit related observations and additional insights from those more deeply familiar with the review experiences that have accrued for both pools and casks during my growing absence from the field of spent fuel criticality safety analysis in recent years.

Where applicable, the discussions refer to physically relevant technical information and insights developed by NMSS, RES, the United States Department of Energy (DOE), the International Atomic Energy Agency and others primarily for the cask storage/aging, transport, and disposal of irradiated light-water reactor (LWR) fuel. Of particular note are (a) studies performed by ORNL and others to support RES and NMSS in the development of review guidance on burnup credit and other analysis issues for evaluating criticality safety in cask systems for spent fuel transport and storage (e.g., References [10] thru [35]), and (b) studies of LWR spent fuel criticality safety analysis and validation performed for the DOE and reviewed by NRC to support cask and repository safety evaluation and licensing (e.g., References [36] thru [44]).

That these discussions primarily address pressurized-water reactor (PWR) fuel rather than boiling-water reactor (BWR) fuel is mainly a reflection of the fact that PWR fuel bundles, with their much larger pin arrays (e.g., 17x17 versus 8x8), inherently accommodate far fewer between-bundle absorber plates per ton of stored fuel. This fact -- along with related technical considerations owing to the far greater variability and complexity of BWR fuel depletion and reactivity phenomena and the resulting practical necessity of using more simplifying modeling conservatisms with implied safety margins in the criticality safety analysis of BWR spent fuel pools -- suggest that the more safety-significant analysis guidance needs and issues will often prove to be those involving PWR fuel. On the other hand, lacking, for example, the safety margins provided by the presence of soluble boron during pool storage, BWR fuel may ultimately warrant closer consideration than indicated in the discussions that follow.

Finally, as noted and summarized above, the discussion of each area closes by giving highly provisional upper and lower estimates of the amounts by which the k_{eff} values calculated for hypothetically identical configurations of irradiated fuel might vary in the absence of adequate unified guidance. It is emphasized that these provisional estimates are intended only for illustrative purposes and may thus warrant eventual replacement by estimates derived more formally from analytical studies compiled to support the establishment of unified guidance.

1. Burnup Credit Isotopics

Here the term “burnup credit” refers to the practice of crediting the reduced reactivity of burned fuel versus fresh fuel when analyzing the criticality safety of spent fuel systems. First among the relevant concepts for burnup credit analysis are the distinctions between analysis models that seek to apply partial credit versus full credit for the calculated reactivity decrements caused by fuel burnup. As illustrated by example in Figure 1, a simple rule of thumb to be noted in this context is the following:

In PWR spent fuel without burnable poisons, roughly two-thirds to three-fourths of the nominally calculated total burnup reactivity decrement can be attributed to changes in actinide isotopic compositions (i.e., more specifically, to the changing concentrations in fuel of heavy metal nuclides, chiefly the thermally fissile actinide nuclides, U-235, Pu-239, and Pu-241, and the predominantly thermal neutron absorbing fissionable actinide nuclides, U-238 and Pu-240). The remaining fraction of the nominally calculated burnup reactivity decrement is then attributed to the many neutron absorbing fission product nuclides that accumulate (and deplete) with fuel burnup.

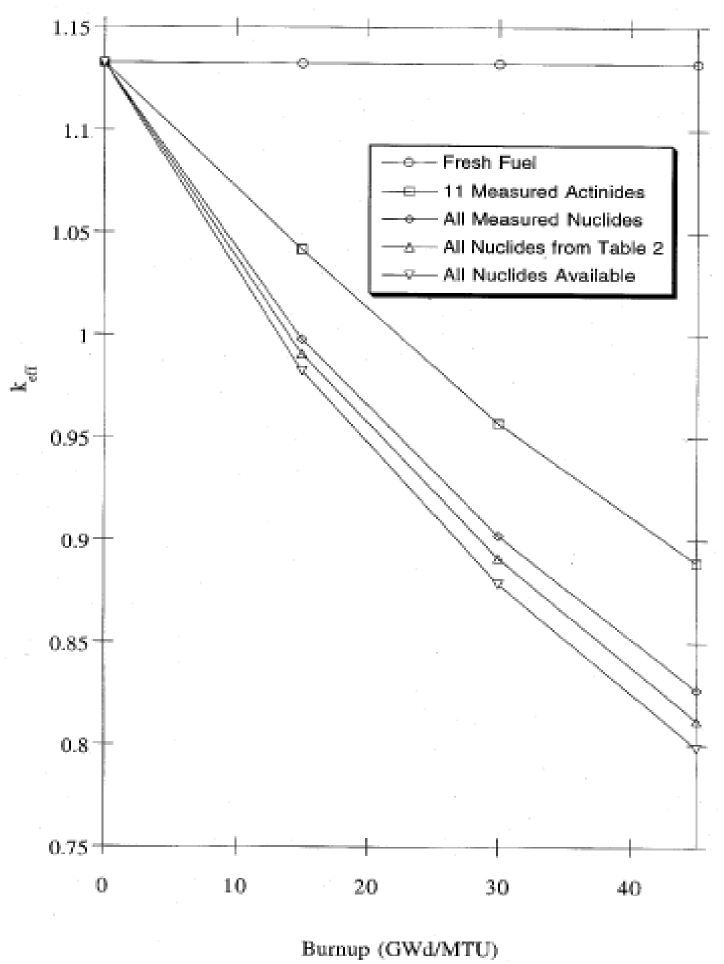

Figure 1. (From NUREG/CR-6665[10]) Values of k_{eff} for a generic rail cask as a function of burnup using different sets of isotopic assumptions and 5-year cooling time

It is instructive to note here that, since about 1980, pool criticality analyses have used full burnup credit in addressing the criticality safety requirements of 10 CFR 50.68 and General Design Criterion 62. As noted in a Regulatory Issue Summary issued by NRR in 2001 [45], initial methods of full burnup credit analysis for pools were based on criticality code models that simply equated spent fuel to fresh fuel with reduced U-235 content. The equivalencing process typically employed simple pin-cell or lattice code models to equate the lattice reactivity of reduced-enrichment fresh fuel to that of spent fuel of a given burnup, whereby the spent fuel lattice was modeled with nominally computed concentrations (i.e., concentrations not adjusted for validation-derived biases and bias uncertainties) of a comprehensive set of actinide nuclides and fission product nuclides. The use of such simple equivalencing methods was discontinued after NUREG/CR-6683 [11] showed them to yield systematically non-conservative results when applied to various analyzed configurations of pool storage.

By contrast, the burnup credit guidance for casks that NMSS developed with support from RES and ORNL starting in 1998 has defined the technical basis for allowing limited actinide-only burnup credit, a form of partial burnup credit that only credits the computed reactivity decrement from major actinides. Even this partial level of cask burnup credit is approved only when subject to the added requirement that residual validation and modeling uncertainties must be estimated and compared for offset against estimated residual reactivity margins implied by the neglect of fission products [14, 46, 47]. Direct credit for neutron absorption by fission products is approved per Part 71 guidance only where justified by the review of additional supporting technical information [47]. The first Part 71 approval of a criticality safety licensing evaluation employing limited credit for fission products as well as actinides was granted in late 2006 [48]. In all cases, the approved modeling methods for transport cask burnup credit incorporate only those nuclides for which measured isotopic assay validation benchmark results have been provided and, when so doing, apply specific adjustments derived from the benchmark results to conservatively address biases and uncertainties in the computed concentrations of the actinide and fission product nuclides included in the model.

Meanwhile, the simple spent-to-fresh-fuel equivalencing methods used early on for storage pools were replaced starting in 2000 by today's more sophisticated methods for full burnup credit in pools, including those that explicitly incorporate all significant actinide nuclides in combination with a simplified approximation of the totality of all fission product nuclides. Specifically, all fission product nuclides are replaced in such models by an "equivalent" artificial concentration of boron-10 in the fuel pellets. The boron-10 equivalencing step generally uses a simple lattice model that contains computed concentrations of all actinide nuclides and all fission product nuclides (e.g., 36 explicit fission product nuclides plus two "lumped" fission products), again without adjusting any of the computed nuclide concentrations for assay-derived validation biases and bias uncertainties. Moreover, this is done with no evident consideration given to the potentially case-dependent distortions introduced by replacing the complex neutron energy dependence of the various fission product nuclide cross sections with the simple "1/v" neutron energy dependence of the boron-10 cross section.

To provide insights into the reactivity effects associated with the distinctions among such approaches to burnup isotopic modeling, the above noted descriptions could be accompanied by sample k_{eff} results calculated for representative burned fuel in generically representative pool rack and/or cask basket configurations.

It is herewith provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 2 to 5 percent due to the absence of unified guidance for this analysis area.

2. Burnup Credit Validation

As evident in the preceding discussion, burnup credit validation is among the more prominent analysis areas to be addressed by unified guidance. The term "validation" refers here to the quantification and treatment of potential biases and uncertainties in the code calculations used for demonstrating criticality safety compliance. Per relevant consensus standards (e.g. the American National Standards Institute / American Nuclear Society standard ANSI/ANS-8.24-2007 [49]), the validation of a criticality analysis tool (i.e., a computer code with well defined input modeling practices) is to be achieved for a given range of uses by benchmarking that tool against integral measurement data from applicable experiments, tests, and/or operations.

Unified guidance for this analysis area should address the graded levels of conservatism implicit in the various approaches to evaluating and treating validation-related uncertainties in conjunction with descriptions of the graded burnup isotopic modeling options alluded to above. Such a discussion could proceed in a conceptual progression like the following:

(1) Nominal full burnup credit with isotopic simplifications: Modeled with nominally calculated explicit isotopic compositions of essentially all actinide nuclides and with the totality of all fission product nuclides approximated as an equivalent concentration of boron-10

(2) Nominal full burnup credit: Modeled with nominally calculated explicit isotopic concentrations of essentially all actinide and fission product nuclides

(3) Best-estimate full burnup credit: Modeled with calculated explicit isotopic concentrations of effectively all actinide and fission product nuclides as adjusted to compensate for estimated or validation-assay-derived calculational biases

(4) Slightly conservative full burnup credit: Modeled with calculated explicit isotopic concentrations of effectively all actinide and fission product nuclides as conservatively adjusted to account for estimated or validation-assay-derived calculational biases and bias uncertainties

(5) Mildly conservative partial burnup credit: Modeled with calculated explicit isotopic concentrations of major actinide nuclides and major fission product nuclides as conservatively adjusted to account for estimated or validation-assay-derived calculational biases and bias uncertainties for each modeled nuclide, whereby the neglect of minor absorber nuclides helps conservatively offset residual uncertainties due to limited or missing validation data

(6) Highly conservative partial burnup credit: Modeled with calculated explicit isotopic concentrations of major actinides and only selected major fission product nuclides as conservatively adjusted to account for validation-assay-derived calculational biases and bias uncertainties for each modeled nuclide, whereby the neglect of some significant absorber nuclides helps conservatively address with higher confidence the residual uncertainties due to limited or missing validation data.

The existing staff guidance for pools includes the following key statement, which was based expert opinion informally elicited in 1998 without the benefit of rigorous validation studies [50]: *"In the absence of any other determination of the depletion uncertainty, an uncertainty* *equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption."*

Appendix A to this paper discusses the potential biases and uncertainties indicated by various rigorous attempts to validate burnup credit analysis tools against the most applicable available data for quantifying the individual and combined biases and bias uncertainties in (a) calculating the burnup-history dependent isotopic concentrations in spent fuel of individual fissile actinides, neutron absorbing actinides, and neutron absorbing fission product nuclides, and (b) calculating the neutron multiplication factors of such spent fuel compositions in representative cask basket configurations.

It is herewith provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 1 to 3 percent due to the absence of unified guidance for this analysis area. Further technical insights pertinent to the establishment of unified guidance for this analysis area are discussed in Appendix A.

3. <u>Rodded Burnup Histories</u>

This analysis area concerns the reactivity increasing effects from increased plutonium production in fuel burned in the presence of inserted control rods, part-length power shaping rods, or removable absorbers such as wet annular burnable absorbers (WABAs), etc. The importance of accounting for such neutron spectral hardening effects in the actinide isotopic calculations for fuel burnup has been as shown in NUREG/CR-6761 [18], as illustrated in Figure 2 for the case of burnup histories in the presence of WABAs, which were widely used in PWR cores through much of the 1980s.

Particularly worth noting in this context is the apparent need to account for similar spectral hardening effects on the actinide isotopics of spent fuel when modeling fuel bundles whose axial burnup profiles suggest that they may have been affected by the insertion or presence of control or power shaping rods. Based on further information in [18], it is herewith provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 1 to 3 percent due to the absence of explicit unified guidance for this analysis area.

4. <u>Spatial Burnup Profiles</u>

This analysis area addresses how the axial and transverse profiles of burnup within fuel bundles affect spent fuel reactivity. Existing review guidance for axial profiles is based largely on NUREG/CR-6801 [23] and seems to have been implemented in a relatively consistent manner for both pools and casks in recent years. As noted above, further guidance may be needed on combining the spectral isotopic and axial burnup shaping effects of rods. Studies of transverse burnup profiles show significant reactivity effects only in small arrays of fuel bundles, like those in truck casks, and may be conservatively modeled using methods and assumptions as described in Reference [51]. It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 1 to 2 percent in the absence of explicit review guidance for this area.

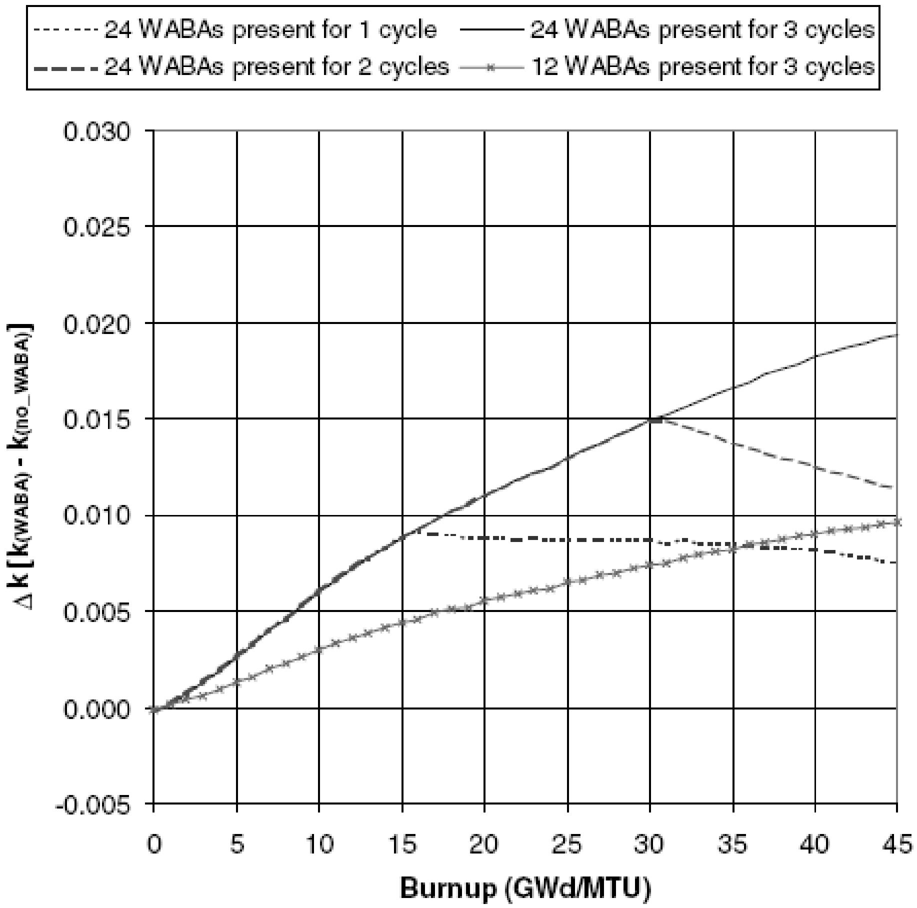

Figure 2. (From NUREG/CR-6761[18]) Δk values as a function of burnup for Westinghouse 17×17 fuel with 3.0 wt % U-235 initial enrichment that has been exposed to Westinghouse WABA rods (3 burnup cycles of 15 gigawatt-days per metric ton uranium (GWd/t) each were assumed)

5. <u>Pin Burnup Modeling</u>

Guidance for this area has been very limited to-date. More explicit guidance would combine and summarize insights from past and supplemental analytical studies on core lattice physics modeling sensitivities and commonly used assumptions and approximations concerning phenomena like the following as they affect fuel pin burnup and reactivity effects in spent fuel:

a. fuel pin radial temperature profiles,

b. radial reaction rate and depletion profiles including the skin or rim effect,

c. uniform and granular integral burnable poisons,

d. burnable poison coatings,

e. fuel pellet-stack thermo-mechanical, material, and dimensional effects, including radial and axial thermal expansion, irradiation swelling or densification, and the potential migration and gap accumulation of mobile neutron absorbing species,

f. cladding thermo-mechanical, material, and dimensional effects, including external (water) and internal (fission gas) pressure loadings with elastic and plastic strain (e.g., creep down), as well as cladding oxidation, crudding, corrosion, and hydriding;

g. intra-assembly, inter-assembly, and by-pass coolant void distributions (BWR),

h. grid spacers,

I. part-length rods, and

j. in-core gamma and neutron detector response models in conjunction with (i) and other key effects above.

It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0.5 to 1.5 percent in the continued absence of explicit review guidance for this area.

6. Spent Fuel Record Accuracy

This analysis area considers the criticality safety implications of the limited and variable accuracy of spent fuel records. Such considerations were discussed at a recent RES seminar [52] and have been the subject of recent work in RES's research program on spent fuel burnup credit. Among the key questions for this analysis area are the following:

- How does the criticality analysis account for the recognized possibility that several adjacent fuel bundles may have actual burnup values several percent lower than recorded?
- How does the analysis account for the lesser accuracy and sophistication of older fuel burnup records, such as the 1960s-70s practice at some plants of assigning all spent fuel bundles in a given batch the batch average value of burnup?
- How reliably are bundle burnup histories with WABAs or other rods inserted (see Rodded Burnup Histories above) captured in the storage record of a fuel bundle?

It is herewith provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0.3 to 1 percent due to the absence of explicit unified guidance for this analysis area.

7. Absorber Plate Granularity

Boral absorber plates have their neutron poison material in the form of relatively coarse grains (e.g. 100 micron) of boron carbide. Neutron transmission tests comparing the effectiveness of Boral plates against ZrB_2 coated aluminum plates have repeatedly shown substantially reduced neutron absorption due to poison granularity effects in Boral. Attributed variously to neutron streaming, neutron channeling, and grain self shielding phenomena, these poison granularity effects are typically shown to equate a Boral plate's effectiveness at attenuating thermalized neutrons to that of a homogeneous ZrB_2 coated plate with 20 percent, or even 35 percent, less boron-10 per unit area [53, 54, 55]. For this reason, the review guidance for criticality safety analysis under Part 71 has historically prescribed the homogeneous computational modeling of absorber plates at no more than 75 percent of their minimum assured poison areal density [56]. On the other hand, the review practices applied to pools have generally allowed intact absorber plates to be homogeneously modeled at 100 percent of their minimum assured poison areal density.

Modern Monte Carlo neutron transport codes like MCNP [57] can explicitly model granular neutron absorbers like Boral by representing the grains as spheres in either regular arrays (sc, bcc, or fcc) or random arrays. My own unpublished modeling studies with such codes have been able to qualitatively reproduce the measured attenuation results noted in the published literature and elsewhere by rigorously modeling the attenuation experiment as a fixed source problem. Moreover, by applying the same Boral plate models to representative criticality eigenvalue problems, my unpublished studies have further shown that granularity generally tends to affect attenuation measurements far more than criticality. For example, an explicit plate model that is only 75 percent effective in a simulated attenuation problem may typically prove to be over 90 percent effective in various simulated criticality problems. A recent published study has shown similar effects [58]. Therefore, the guidance options for modeling Boral plates in criticality analysis might include something like the following:

(a) Qualify an explicit poison granularity model by using it in the benchmark simulation of an actual attenuation measurement setup (e.g., rigorously modeling the fixed Cf source in water, test plates, and neutron detector) and comparing the computed versus measured results. The simulation results should conservatively reproduce the measured ones by showing the modeled plate to be no more than 90 or 95 percent as effective as measured (e.g., 72 or 76 percent effective modeled versus 80 percent effective measured).

(b) Apply the mildly conservative plate granularity model thus qualified to the criticality problems of interest.

It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0.3 to 1 percent in the continued absence of explicit review guidance for this area.

8. Bundles with Removed Pins

Sample pool licensing submittals have included analyses for the controlled storage of loose fuel pins but not for any assemblies from which any pins had been removed. Based on analysis results in NUREG/CR-6835 [25], it is provisionally estimated that accounting for the reactivity increasing effects of removing fuel pins (typically maximized at ~20-40 removed pins) from PWR fuel assemblies increases predicted pool k_{eff} by 0.5 to 1.5 percent and that the k_{eff} values calculated for hypothetically identical configurations of burned fuel with removed pins might vary accordingly in the absence of explicit review guidance for this area.

9. Cooling Time Am-241 Credit

Also apparently lacking is explicit guidance on the uncertainty issues that arise when considering the crediting of neutron absorption by Am-241 when taking credit for the long term cooling time effects of Pu-241 decay in spent fuel. Of particular concern here are the lack of criticality validation experiments that include significant resolved Am-241 absorption effects, the quality of evaluated absorption cross section data for Am-241, and the potential for non-conservative long term effects resulting from isotopic biases and depletion models that are conservative over the short term. Therefore, conservative limits on the credit allowed for Am-241 may be warranted. NUREG/CR-6781 and NUREG/CR-6979 provide useful insights on this topic [20, 29]. It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0 to 0.5 percent in the continued absence of explicit review guidance for this area.

10. Actual Fuel Pin Conditions

Per existing review guidance, the modeled spent fuel pellet and cladding dimensions and cladding compositions are those of fresh fuel. No evident consideration is given to how pool reactivity may be affected by the actual conditions of spent fuel such as those associated with pellet swelling, cladding creep down, cladding hydriding via the oxidized layer in high burnup fuel, water inleakage through pinholes and cracks, and axial growth of the active fuel region. Unpublished studies suggest that the maximum neutron moderation effects from the hydrogen in hydrided high burnup cladding can be represented conservatively by simply modeling water in the pristine pellet-clad gap. It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0.2 to 0.5 percent in the absence of explicit review guidance for this area.

11. Monte Carlo Undersampling

There is a clear lack of pool review guidance on considering the computational issues of Monte Carlo source under-sampling and false convergence. Such issues can lead to the potential for gross under-predictions of k_{eff} when analyzing so-called "k_{eff} of the world" problems, including for example (a) large pool arrays in which the global k_{eff} is dominated by small regions of higher local reactivity, (b) potentially peaked local reactivity effects near pool walls [59], and (c) effects of any local gaps in absorber plate coverage [20]. It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0 to 1 percent in the absence of explicit review guidance for this area.

12. Wall Reflection Effects

Review guidance may be needed for considering potential configurations where neutron moderation and reflection by the water and pool or cask wall materials just outside a real (i.e., finite) fuel rack or basket array can yield k_{eff} values higher, for example, than the k_{eff} values computed for an infinite array of rack or basket cells. A stylized storage array problem suggesting the potential importance of carefully accounting for such wall effects is included in Reference [59]. This analysis area may be particularly relevant for pool analyses where it is common practice to model infinite rack arrays in lieu of explicit finite arrays.

The potential for such wall effects to exist in real spent fuel configurations may be heightened by the fact that neutron poison plates are often omitted by design from the outer walls of rack and basket arrays. A preliminary understanding of how a net positive reactivity effect (herein called "wall reflection reactivity effect" for brevity) can arise in such cases may then be gained by noting that an idealized (i.e., purely mathematical, non-physical) mirror reflection of neutrons at an outer rack wall clearly would produce the reactivity effect of two mirror-image racks or, equivalently, a double-size rack in which poison plates are absent from the cell walls at the centerline of symmetry. While such an idealized, leakage-free, mirror reflection of neutrons is in fact not physically possible, it is nevertheless conceivable that the real physics of neutron scattering (i.e., combined reflection and moderation) in adjacent water and pool wall materials could produce analogous, albeit less pronounced, positive reactivity effects in real storage pool configurations. This is further explained in the next paragraph.

For such configurations, the k_{eff} of the real finite rack will exceed the k_{eff} computed for an infinite array of rack cells whenever the reactivity-decreasing effects of net neutron leakage are exceeded by the reactivity-increasing spectral effects of neutron moderation and reflection by the water and/or wall materials just outside the poison-plate-free outer rack walls. [Describing these effects using the physical terms of the famed six-factor formula, one would find in such cases that, in relation to the case of an infinite array, the neutron scattering effects in the water and/or wall materials just outside the poison-plate-free outer rack walls are producing a positive change in the product of the four spectral factors (i.e., the fast-fission factor, the resonance escape probability, the thermal utilization factor, and the reproduction factor) that exceeds the concurrent negative change in the product of the two non-leakage factors (i.e., the fast and thermal non-leakage probabilities), the latter product generally becoming <1 in going from an infinite to a finite array.]

I am not aware of any published studies that investigate the potential for such wall reflection reactivity effects to exist in representative pool or cask configurations. Should an exhaustive literature search fail to find such studies, it is recommend that simple analytical modeling studies be performed to determine whether net positive pool wall reactivity effects are likely or possible in practice and, if they are indeed likely or possible, whether they are potentially large enough to merit explicit consideration for evaluating criticality safety. For now, it is provisionally estimated that correcting an infinite array analysis to realistically address such effects could eventually increase the predicted values of system k_{eff} by 0 to 1 percent.

13. Offsetting Conservatisms

The existence of offsetting modeling conservatisms may be used to rationalize the acceptability of various other modeling practices and uncertainty treatments. However, care should be given to ensuring that the reactivity effects of the modeling conservatisms are large enough to offset the other concerns. Therefore, the reactivity effects of commonly used modeling conservatisms should be analyzed for representative configurations. Such modeling conservatisms might include, for example, the following:

- Practices for approximating the pin-by-pin variation of burnup and burned fuel compositions within a fuel bundle at a given elevation as a single average pin burnup and composition.
- The practice of modeling the unenriched or depleted axial blankets in a fuel bundle as active fuel.
- The assumption of conservative fuel depletion parameters such as moderator temperature, soluble boron history, fuel temperature, power density, etc.

It is provisionally estimated that the k_{eff} values calculated for hypothetically identical configurations of burned fuel might vary by 0 to 2 percent in the absence of useful review guidance for this area.

Considerations for Establishing and Applying Consolidated Review Guidance

Further insights toward domain-specific applications of consolidated review guidance on spent fuel criticality analysis may be gained by considering the respective review practices and staff resources typically applied by NRR versus NMSS to the criticality safety evaluations for pools versus casks. For example:

- Each year, NRR and NMSS generally seem to undertake similar numbers of criticality safety licensing review cases for pools and casks, respectively. Given the more extensive use of burnup credit in pools, the criticality analyses reviewed by NRR would seem in most cases to be fundamentally more complex than those reviewed by NMSS.

- NMSS typically allocates 5 to 6 full-time equivalents (FTEs) of fully specialized staff effort to its licensing evaluations of criticality safety for spent fuel casks. Such reviews in NMSS very often include the performance of independent confirmatory calculations.
- NRR, on the other hand, seems to allocate about 1 or 2 FTEs of fully specialized staff effort to its licensing evaluations of criticality safety for fuel storage pools. NRR's pool criticality reviews do not routinely include the performance of independent confirmatory calculations.
- While the lower levels of review effort typically allocated by NRR would seem to suggest a greater need for explicit guidance, one nevertheless finds that the NMSS guidance is in fact more explicit and rigorous than the guidance used to-date by NRR.

Related regulatory and risk considerations for criticality safety in pools, casks, and other systems may involve questions such as the following:

(1) Should the NRC consider risk-informed modifications or exceptions to its current strategic goals regarding criticality safety? Event risk is the product of event frequency and event consequences. The current safety goal calls for complete success at preventing inadvertent criticalities, i.e., an event frequency approaching zero. The basic principles of risk-informed regulation suggest that the importance of preventing an event is proportional to its potential adverse consequences. Relevant questions may thus include:

   a. What would be the expected or potential consequences of a spent fuel criticality event in (a) a storage pool, (b) a shipping cask, (c) a repository system, or (d) a reprocessing system in terms of:

      - Damage to health and safety of workers and the public, etc. (low to moderate)?
      - Loss of public confidence, political/media outrage, etc. (high to very high)?

   b. Are criticality accidents acceptable where it can be shown with reasonable assurance that they will not cause severe damage to worker or public health?

      - If not acceptable, then the emphasis on criticality prevention is indeed warranted.
      - If acceptable, then less emphasis on prevention may be called for.

   c. Is it appropriate to risk-inform the consideration of spent fuel criticality safety issues solely by evaluating the product of event frequencies and potential health consequences? Should we also consider criticality accident consequences in terms of lost public confidence and the resulting potential to unduly limit the future benefits of nuclear energy?

(2) What are the safety and risk implications of reduced safety margins or eventual regulatory non-compliance in various spent fuel systems? Consider pools for example:

   a. What would be the safety and risk implications for the hypothetical case of a storage pool where k_{eff} may be 0.03 higher than the <0.95 compliance criterion in 50.68(b)(4)? Note that the unborated k_{eff} <1.0 compliance criterion in 50.68(b)(4) has no explicit margin.

   b. What are the difficulties in meeting the alternate criticality accident alarm requirements of 50.68(a) for storage pools?

   c. Should additional controls be placed on pool boron concentration to reflect an eventually heightened reliance on boron for preventing criticality?

Appendix A

Needs for Unified Guidance on Burnup Credit Validation

Background

The need to validate analysis codes takes on special significance wherever criticality codes are used to demonstrate the adequate prevention of self-sustaining fission chain reactions outside reactors. Whereas reactors are equipped with means for confirming shutdown margin and monitoring subcritical neutron multiplication on controlled approach to critical, no such means are generally available for confirming or monitoring subcritical safety margins for fissionable materials outside reactors.[1] Thus lacking measured confirmation, nuclear criticality safety analysts must instead rely solely on the prior validation of the codes they use to predict subcritical safety margins.

Criticality safety code validation is therefore done for the highly specific purpose of determining what adjustments must be made to code calculated results in order to yield final predictions that, with reasonable confidence, will not significantly underestimate the actual k_{eff} values in a given system.

Introduction

The validation of criticality codes for use on systems with fissionable materials other than spent fuel is generally based on rigorous benchmarking against various applicably similar laboratory critical experiments [49]. To achieve comparable validation rigor and confidence in the criticality safety analysis of spent fuel systems, one would therefore ideally wish to benchmark the codes against laboratory critical experiments that feature various applicably similar configurations of spent fuel.

Unfortunately, no laboratory critical experiments have ever been performed with fully applicable quantities of spent fuel. Although proposed (e.g., per Reference [60]), even partially applicable benchmark experiments with relatively limited amounts of spent fuel material have remained elusive due in part to the necessity of adding costly shielding and confinement measures to existing experimental facilities to enable their safe handling of spent fuel. Similarly, proposals to perform fully applicable source-driven subcritical benchmark experiments in PWR spent fuel storage pools still wait to be pursued for reasons ranging from lacking support for enabling R&D to anticipated hindrances in securing a welcoming pool owner.

Lacking the desired criticality benchmark experiments with spent fuel, validation efforts undertaken for the spent fuel transport and disposal sectors have thus had to rely on benchmarking against less suitable combinations of available data from laboratory critical experiments, spent fuel destructive assays, and commercial PWR restart criticals. Pending efforts in this area are presently focused on incrementally supplementing the existing critical experiment database with new critical experiments that contain test samples of individual fission product nuclides.

Burnup Credit Validation Results, Issues, and Insights

Burnup credit validation approaches accepted within the limited context of actinide-only burnup credit [46] have entailed combining the separate results from code computed versus measured benchmarks on:

- Critical experiments featuring various configurations of unirradiated low-enriched uranium (LEU) and mixed U-Pu oxide (MOX) fuel materials, and
- Spent fuel isotopic assays for the concentrations of the same actinide (i.e., U, Pu, Am) isotopes present in the LEU and MOX critical experiments.

Moving beyond actinide-only burnup credit requires additional validation for both the computed accumulation and the computed reactivity effects of fission products. Of the many significant neutron absorbing fission product nuclides, only a few have been included significantly in spent fuel assay data and fewer yet have had their reactivity effects measured in critical experiments. Validation data for extended or full burnup credit must therefore come from PWR restart criticals.

Results of rigorous MCNP/SAS2H [57, 61] benchmark analyses on 45 PWR restart criticals [37, 38, 39, 40, 42] have revealed substantially non-conservative k_{eff} computational bias trends that increase with core average burnup.[2] From the k_{eff} regression analysis of these results shown in Figure A-1, it is readily seen that a bias and uncertainty adjustment of at least 2 percent would have to be applied when, for example, hypothetically using such MCNP/SASH code models to predict the 95/95 subcritical limit in a similar PWR core at a time when, late-in-cycle, the core-average fuel burnup is 34 GWd/t.[3]

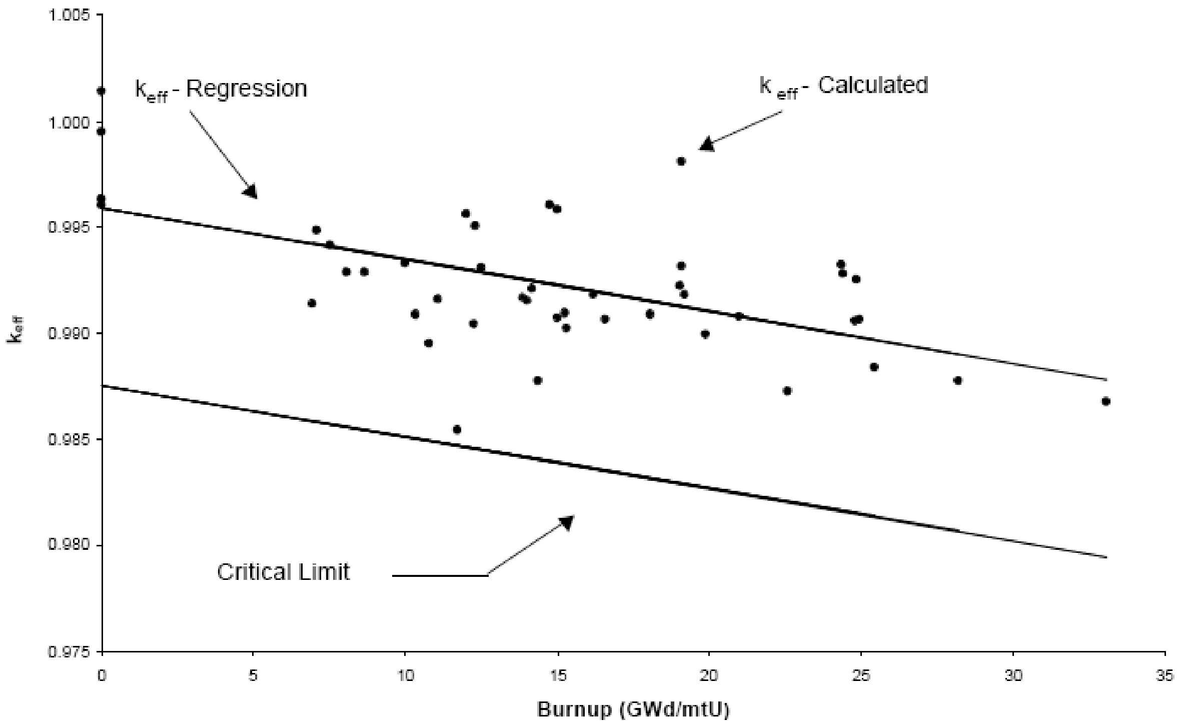

Figure A-1. (Copied from [38]) Results of MCNP/SAS2H benchmark analyses on 45 PWR restart criticals at various core average burnups

Bias and uncertainty allowances significantly greater than 2 percent would then clearly be indicated by any attempt to extrapolate these PWR restart benchmark results to validate the use of similar MCNP/SAS2H models to predict 95/95 subcritical limits for spent fuel in storage pool or cask configurations. The basis for this assertion and its importance will be further discussed in subsequent paragraphs.

From the calculated reactivity decrement curves shown in Figure A-2, one can readily determine for example that, for loading PWR fuel burned to 40 GWd/t into configuration neutronically similar to the generic burnup credit (GBC) cask model [14], an uncertainty assumption of 5 percent of the burnup reactivity decrement would amount to only 1.2 or 1.4 percent $\Delta k_{eff}/k_{eff}$ for fuel initially enriched to 4.5 or 3.0 weight percent ^{235}U, respectively. Although burnup reactivity decrements computed for pool racks are not shown here, they would generally be comparable to those shown for the GBC cask models and clearly less than those shown for the pin cell models [51].

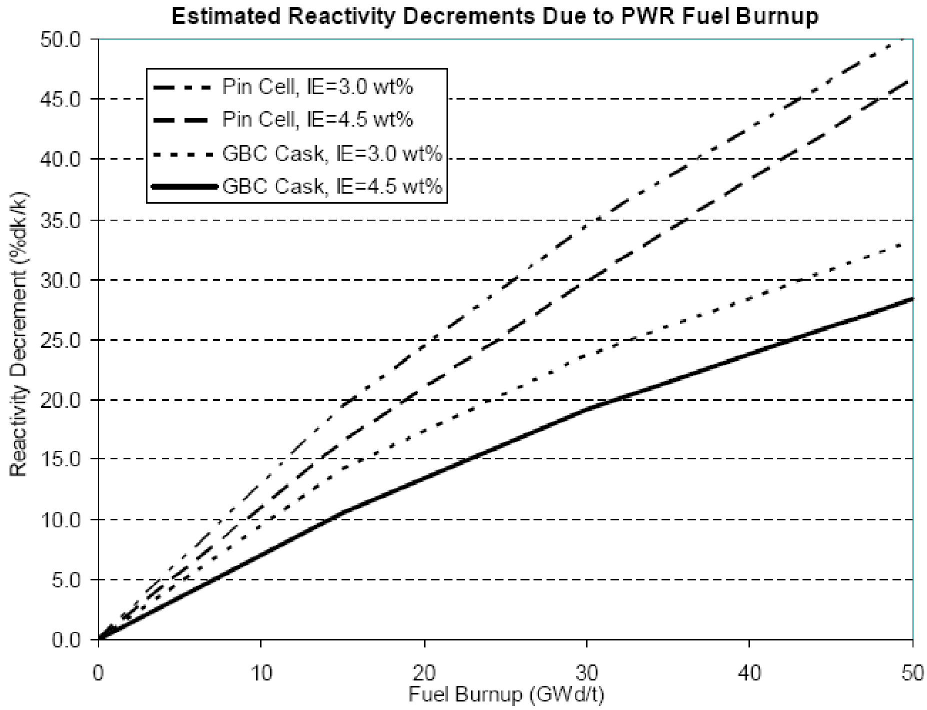

Figure A-2. Estimated burnup reactivity decrements for PWR fuel initially enriched (IE) to 3.0 and 4.5 weight percent ^{235}U as computed with all actinide and fission product nuclides in (a) a simple pin cell model [51] and (b) a generic burnup credit (GBC) cask model [14]

It is important to acknowledge here that the technical applicability and adequacy of PWR restart criticals for validating the analysis of PWR spent fuel burnup credit has long been a subject of discussion and debate within the international criticality safety analysis communities for spent fuel in pools and casks. In fact, one of the latest products from RES's ongoing burnup credit research program at ORNL is a report [27] that employs advanced sensitivity and uncertainty analysis methods to better understand precisely these issues of applicability and adequacy. However, in this case -- where the benchmark analysis of many PWR restart criticals using reference-quality criticality code models has indicated substantially non-conservative bias and uncertainty trends – it is clear that we as regulators should take the more conservative position of assuming applicability to the extent that the results may suggest a need to proceed with more caution than heretofore applied to the analysis of spent fuel pool criticality safety.

Also important to understand in this context is the frequently cited fact that the analysis tools used by licensees for core reload, core follow, and restart criticality have long been able to accurately predict PWR restart critical k_{eff} to within well less than 1 percent or 50 ppm soluble boron. When considering any claims about how this fact might relate to the expected accuracy of such tools, or even other tools like MCNP/SAS2H, when they are applied to spent fuel criticality safety analysis, it is important to weigh relevant observations like those noted and discussed below [62]:

- First to be noted is the fact that many reactor years of PWR restart experience have effectively taught licensees how to "tune" and "calibrate" their codes, input models, and bias-adjusted outputs to enable consistently accurate predictions of PWR restart criticality. Such tuning and calibration relies on highly specific combinations of coded physics formulations (e.g., cross section energy group structures, evaluated nuclear data files, resonance approximations, self-shielding treatments, scattering kinematics treatments, 2D transport theory, 3D transport or diffusion theory, etc.) and input modeling practices (e.g., geometry simplifications and approximations, material geometry meshing or noding, spectral noding, temperature noding, boundary conditions, depletion time stepping, spectral time stepping, explicit versus lumped nuclide tracking, etc.).

- The preceding observation suggests that a careful discussion may be warranted on exactly how the PWR restart prediction accuracies made possible by the tuning and calibration of restart analysis tools against historical PWR restart data might eventually be made to extrapolate and propagate to the criticality analysis of spent fuel in storage pools. Even assuming that the same codes were eventually to be used with consistent modeling practices where possible, such a discussion may lead one inevitably to conclude that the necessary extrapolation and propagation processes would, at best, be fraught with complexity, needs for more information, and various added uncertainties that would be hard to quantify.
- Underlying the many practical difficulties inherent in any attempt to extrapolate and propagate the tuned and calibrated accuracy of licensee restart analysis tools into the realm of spent fuel burnup credit analysis are the related and previously mentioned issues of validation applicability and adequacy, which focus largely on understanding and quantifying the ways in which PWR restart cores differ neutronically from PWR spent fuel storage systems.

Finally, as noted in a February 2008 workshop presentation by ORNL [63], it is important to recognize that not all historically accepted burnup credit validation practices are fully consistent with relevant consensus standards. The establishment of unified guidance on burnup credit validation would serve to bring accepted practices in this area into more consistent alignment with such standards.

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[1] Note that criticality accident alarm systems, which are required at pools by 50.68(a) and 70.24 as an alternative to meeting the preventive requirements of 50.68(b), are generally not designed to monitor subcritical margins or detect impending criticality but more simply to detect a criticality event wherever it occurs in the system and quickly trigger mitigative actions to minimize dose consequences to nearby workers and others.

[2] The 45 restart criticals consisted of zero-power critical conditions measured at two Babcock & Wilcox PWRs and two Westinghouse PWRs during various 12-, 18-, and 24-month refueling cycles in which the loaded fuels had initial enrichments ranging between 1.93 and 4.17 weight percent U-235. Core critical conditions were measured for restarts at beginning-of-life (BOL) with all fresh fuel, at beginning-of-cycle (BOC) with a mix of fresh and burned fuel, and at middle-of-cycle to near end-of-cycle (EOC) with all fuel at least partially burned, whereby core soluble boron concentrations ranged from less than 400 ppm near EOC to over 2300 ppm at BOL.

[3] Note that using simple process-of-elimination logic to connect these restart benchmark results to the results of parallel SAS2H spent fuel assay benchmarks [51, 41, 42, 43] and MCNP laboratory critical experiment benchmarks [40] would seem to suggest that these substantial restart bias trends may reflect a propensity of the MCNP/SAS2H models to substantially over-predict the total absorption of neutrons by fission products. However, no examinations of this or any other diagnostic hypotheses have been published to-date. It has more recently been noted that such trends were not readily apparent in the results of subsequent benchmark studies that used two-dimensional lattice depletion codes in place of the one-dimensional SAS2H code [48]. This observation, however, may need to be reconciled with the fact that code-to-code-to-data benchmark comparisons between SAS2H and the newer two-dimensional TRITON code have shown no substantial differences in the computed concentrations of any significant actinide or fission product nuclides [32].

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