Validation Issues for Depletion and Criticality Analysis in Burnup Credit

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VALIDATION ISSUES FOR DEPLETION AND CRITICALITY ANALYSIS IN BURNUP CREDIT*

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Abstract

This paper reviews validation issues associated with implementation of burnup credit in transport, dry storage, and disposal. The issues discussed are ones that have been identified by one or more constituents of the United States technical community (national laboratories, licensees, and regulators) that have been exploring the use of burnup credit. There is not necessarily agreement on the importance of the various issues, which sometimes is what creates the issue. The broad issues relate to the paucity of available experimental data (radiochemical assays and critical experiments) covering the full range and characteristics of spent nuclear fuel in away-from-reactor systems. The paper will also introduce recent efforts initiated at Oak Ridge National Laboratory (ORNL) to provide technical information that can help better assess the value of different experiments. The focus of the paper is on experience with validation issues related to use of burnup credit for transport and dry storage applications.

1. INTRODUCTION

Requirements applied within the United States, for validation of codes and data used for criticality safety outside reactors, are provided by ANSI/ANS-8.1 [1]. This standard requires that the calculational method be validated by comparison with "the results of critical and exponential experiments." Such a comparison yields information on biases and uncertainties in the calculational methods and model. The area of applicability for the calculational method is established by the characteristics of the measured critical experiments that are considered in the validation. The standard gives no guidance on how to establish the area of applicability (e.g., which parameters, characteristics, etc., and how similar they should be to the application).

The process of performing criticality calculations for spent fuel in a burnup credit model for transport or dry cask storage requires two distinct sets of calculations — the first to estimate the isotopic contents of spent fuel based on depletion calculations; the second to perform a criticality calculation based on the predicted isotopic contents from the first set of calculations. Thus, application of ANSI/ANS-8.1 to burnup credit validation becomes somewhat complicated by (1) the need to consider both the depletion analysis methodology and the criticality analysis methodology and (2) the lack of spent fuel critical experiments.

The objective of a validation effort per ANSI/ANS-8.1 guidance is to establish a limit for the calculated neutron multiplication factor (k_{eff}) below which the system of concern would be considered subcritical. The "fresh fuel" assumption has provided a simple, bounding approach which allows less scrutiny of the validation needs relative to fuel composition. Under burnup credit, applicability of experiments are not as obvious and validation efforts may be more closely scrutinized to ensure adequate definition and understanding of the subcritical margin.

The nature of experimental data appropriate for use in validation of burnup credit analysis methodologies and the value and applicability of such data have been debated topics for over a decade. Available (albeit some are proprietary) experimental data include chemical assays of spent nuclear fuel (SNF) inventories, critical experiments performed with fresh fuel (unirradiated fissile material) in cask-like geometries, reactivity-worth measurements, subcritical experiments, and reactor critical configurations. The following subsections discuss each of these sources of measured information and their potential value to the validation process.

2. CHEMICAL ASSAY MEASUREMENTS

2.1. Review of Available Data

Radiochemical assay measurements in the United States have been made for select spent fuel nuclides, for both PWR [2 to 8] and BWR fuels [9,10]. In addition, Ref. [11] is a compilation of sources of radiochemical assay data from these and other sources. Reference [12] describes sources for additional isotopic assays and assesses the completeness of available data describing each set of measurements.

Within the United States, chemical assay data have historically focused on the major actinides within PWR spent fuel. The actinides of importance in burnup credit have been measured for approximately 50 PWR fuel samples that provide the basis for performing code validation. Of these 50 samples, only seven had burnable poison rods available during irradiation – an indication of the age of the fuel designs and the data. For most fission product nuclides important in burnup credit, very few assay measurements (typically three samples) have been made in the United States. The enrichment and burnup ranges of the PWR spent fuel samples readily available in the United States are shown in Fig. 1 as the "existing database." The majority of these measurements have been used to determine the biases and uncertainties of computational methods [13 to 15]. With the current trend towards higher enrichment and burnup values, the acquisition of additional assay data to support code validation in this regime is considered a high priority in the United States. Additional PWR and BWR spent fuel assays with the desired characteristics are currently being performed to support U.S. Department of Energy (DOE) programs, but will not be available until sometime in 2001. Other sources of chemical assay data currently exist and/or are planned, largely within programs organized by other countries and held proprietary by those who procured the data. The French program on burnup credit [16], the REBUS program organized by Belgonucleaire [17], the LWR-PROTEUS program organized in Switzerland [18], and referenced Japanese data [19] are all potential sources of additional chemical assay data for use in validation. Figure 1 highlights the characteristics of the known sources of assay data identified by ORNL for potential use in burnup credit validation.

2.2. Validation Approaches

The use of the chemical assays in the validation process involves a comparison of predicted nuclide concentrations to the measured concentrations. The depletion model is based on the known in-core history for the fuel sample that was characterized. Given a significant number of comparisons, it becomes possible to statistically estimate the bias and uncertainty in the calculated prediction of each individual nuclide concentration. The bias is obtained by finding the average difference between computed and measured concentrations for each individual nuclide; the uncertainty characterizes in a statistical manner the variation of individual comparisons around the mean [20]. The total uncertainty should also include statistical uncertainty based on a limited sample size. Reference [20] describes an approach for calculating bias and uncertainties such that one has a reasonable confidence that one can conservatively predict the concentration of a nuclide. Conservatism is defined in terms of correcting a nuclide concentration in such a way that has the effect of maximizing k_{eff} for a system. A second statistical approach is presented in Ref. 21. In both of these procedures, total calculated biases and uncertainties include any biases and uncertainties inherent in the

experimental measurements. Thus, there is potential for offsetting errors in the bias, and the uncertainty may not be properly characterized. However, this is a random process, and non-offsetting errors would also be present.
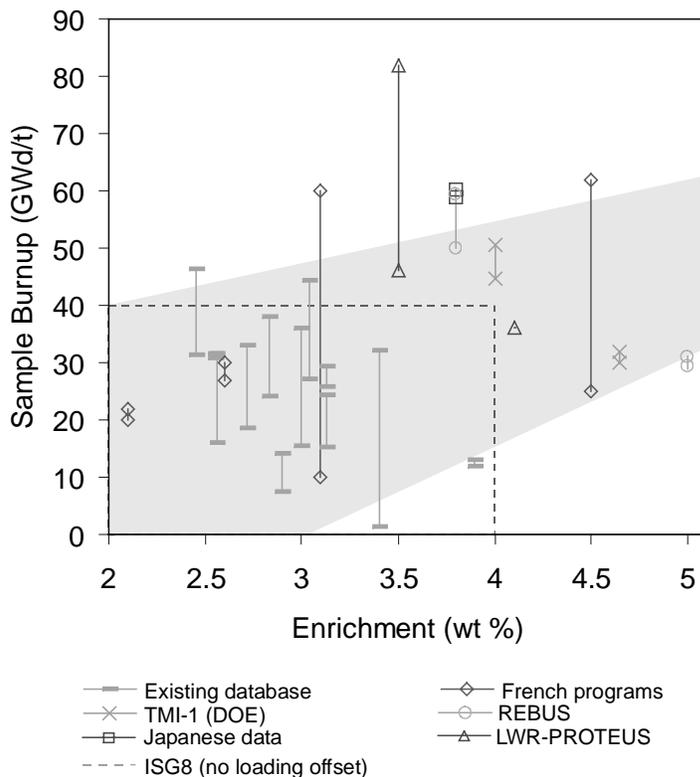

FIG. 1. *Available and Potential PWR Chemical Assay Data Overlaid on a Shaded Range of High Applicability With Respect to the Enrichment and Burnup Regime of Existing Assay Data*

Note that the procedure described above determines the calculational biases and uncertainties for each individual isotope evaluated. Simultaneous application of conservative corrections to individual nuclides within a predicted SNF inventory is a bounding, but unrealistic approach.

Another approach that could be used to obtain uncertainties in the SNF inventory would be to assess the integral effect on k_{eff} due to random variations to the SNF nuclide set within the characterized uncertainty bounds defined for each nuclide. This random variation of the inventory may provide a more realistic distribution of k_{eff} values for the application that can be directly tied to nuclide uncertainties and prevent simultaneous conservative correction of each nuclide. A conservative margin can be assigned based on the expected statistical distribution of k_{eff} values. Perceived advantages (better estimates of the impact of the uncertainty in the spent fuel inventory) and disadvantages (increased complexity and computational time) of this approach are being studied at ORNL [22].

2.3. Extending the Use of the Data

Given the limited number of chemical assays available, and the range of enrichments and burnups represented by these data, it has not been possible to clearly establish trends in biases and uncertainties as a function of the governing parameters. Although some chemical assay data exist for a moderate range of burnups, other factors also vary (i.e., assembly design, operating history, poison concentrations, etc.).

Insufficient data prevent application of a reliable multivariate evaluation. Although additional measurements should be pursued where essential (high-enrichment and high-burnup conditions), the lack of experimental facilities available in the United States to handle and process commercial spent fuel, combined with the cost of the procedure itself, will limit the number of samples available for validation in the near future.

Thus, other options that can provide technical justification for extending the range of the validation and/or interpolating on the range are being considered in conjunction with new experimental data. The dashed lines in Fig. 1 represent the limits beyond which an additional reactivity margin is recommended by the current burnup credit regulatory guidance [23] on transport and dry storage. The recommended reactivity margin is equal to 1 GWd/t for every 0.1 wt % initial enrichment above 4.0 wt %. Developing validation techniques and/or data that enables extension beyond these limits is of keen interest in the United States. However, extending the area of applicability by making use of trends in the bias and uncertainty has proven to be challenging due to a relatively large variability in the existing data and the many factors that may influence the overall bias and uncertainty: fuel enrichment, burnup, assembly design complexity, calculational methods, nuclear data, and uncertainties in reactor operating conditions, irradiation history, and sample burnup. A reliable trending assessment is challenged by the limited amount of experimental data and the large number of different parameters that can affect the bias.

Sensitivity-based methods have been applied at ORNL to assess the influence of nuclear data bias and uncertainties on the isotopic compositions and the k_{eff} of a spent fuel storage cask. These studies indicate that there is a strong correlation between spent fuel systems with a constant enrichment-to-burnup ratio. An example of the band of "high applicability" with respect to the enrichment and burnup regime of existing assay data found by the sensitivity-based methods is shown by the shaded area in Figure 1. The results suggest that existing isotopic assay data may be highly applicable to regimes well beyond that of the measured data. However, there is currently insufficient experimental data to validate these findings. It is anticipated that, as new assay data become available, it will be possible to combine the limited amount of experimental data with the sensitivity-based methods to provide additional evidence to support predictions on bias and uncertainty beyond the range where the majority of experimental data exist.

Reactor critical configurations and some planned reactivity-worth experiments with spent fuel are integral experiments that require prediction of the nuclide composition and k_{eff} analysis. Thus, these experiments are also potential sources of experimental data that may be used to supplement, or potentially replace, the use of assay data. These other data sources will continue to be considered as work continues to expand the range of data available for validation of the SNF nuclide compositions.

3. CRITICAL EXPERIMENTS WITH FRESH FUEL

The validation of criticality safety analysis codes using critical experiments with unirradiated fissile material is a procedure that has been applied for years to meet the requirement of ANSI/ANS-8.1. Experiments exist for a wide variety of conditions representative of pin lattices within cask environments [24]. The value of fresh fuel critical experiments to validation of spent fuel in casks relates to the fact that these experiments provide validation of the particle transport models and cross-section data within cask-like conditions (e.g., similar geometry, reflectors, intermediate absorbers). However, these experiments do not contain the same relative compositions or even all of the nuclides that are present in spent fuel. Thus, there is a need to limit the applicability of the experiments to validation for nuclide compositions present in the critical experiments (typically plutonium and uranium isotopes only) and/or demonstrate their applicability to a spent fuel inventory.

The current U.S. regulatory guidance for burnup credit application to transport and dry storage [23] recommends that burnup credit be limited to the actinide-only nuclide compositions in the SNF. Thus, initial work [21] has placed a strong reliance on mixed-oxide (MOX) critical experiments. However, MOX critical experiments available in the U.S. do not have the same relative composition of uranium and plutonium

nuclides as found in typical spent fuel. Proprietary fresh fuel experiments using pin lattices with uranium and plutonium compositions similar to that for typical spent fuel (37.5 GWd/t) have been performed in France [25]. Approximately 150 experiments with these "HTC" pins simulating SNF compositions have been performed including arrays in pure water, arrays in poisoned water, pool storage configurations, consolidated pool storage configurations, shipping cask configurations, and mixed arrays of HTC and UO_2 pins. Also, proprietary fresh fuel experiments with pin lattices surrounding cans of fission product solutions have been performed in France [26] and would be excellent experiments to use if credit for fission products is sought.

Sensitivity and uncertainty (S/U) methods recently developed [27] at ORNL are being used to provide information on the strengths and potential limitations of various types of experiments relative to validation needs for burnup credit. The S/U methodology utilizes two different parameters as measures of applicability; one is a global measure for system-to-system applicability (c_k value), the other is a nuclide-specific measure of applicability (T value). This S/U methodology has been used to study the applicability of the available U.S. MOX experiments, the French HTC experiments, and the fission product solution experiments. A representative set of approximately 450 UO_2 and MOX fresh fuel criticals have been used in the study together with a representative set of five of the HTC experiments (lattices in pure water) and a single fission product solution (performed with ^{149}Sm) experiment which is publicly available [25].

A set of 36 U.S. MOX experiments proposed [21] for validation of actinide-only burnup credit was analyzed for applicability to a series of pin-cell spent fuel calculations covering burnups of 10-60 GWd/t. The global applicability parameter indicated that 15 of the 36 systems were of value in validating systems with enrichments near 60 GWd/t. For the lower-burnup cases, the number of systems predicted to be applicable fell to 1 and 0 for 30 and 10 GWd/t, respectively. The reduced applicability for lower-burnup values is not surprising since these MOX systems are designed to mockup systems with substantial amounts of plutonium. The standard UO_2 experiments are expected to fill the gap for lower enrichments. The nuclide-specific values indicated the same applicability trends for the primary plutonium isotopes with 0, 2–7, and 7–21 systems meeting the criterion for 10, 30, and 60 GWd/t systems. Thus, these systems are useful for validation of burnup credit studies; however, they must be supplemented with additional systems at low burnups.

The c_k values for the HTC experiments indicate a high degree of applicability to a series of infinite pin-cell calculations for burnups ranging from 10–60 GWd/t. The T values also indicate a high degree of applicability for the primary plutonium isotopes for burnups less than 60 GWd/t. Thus, these experiments are beneficial to burnup credit validation efforts.

The fission product solution experiments have been evaluated using only the nuclide-specific T parameters. This is because the system-to-system parameters are not currently appropriate for fission products due to the lack of uncertainty data on the fission product cross sections. Also an examination of the T values is performed only for ^{149}Sm, since this is the only experiment in the open literature. The T values obtained for this experiment indicated that it is highly applicable to ^{149}Sm capture in the series of pin-cell applications for 10–60 GWd/t. This indicates that the fission product solution experiments should be good experiments for validation of the fission products in a cask environment. These fission product solution experiments are valuable in that they allow for the effect of individual fission product cross-section uncertainties on the system k_{eff} to be evaluated separately.

Work is continuing at ORNL to develop a sound basis for the S/U methodology such that it can be used to enable the maximum benefit to be obtained from validation efforts with well-defined fresh-fuel critical experiments.

4. REACTIVITY WORTH EXPERIMENTS

To bypass the difficulties associated with using spent fuel assemblies in critical experiments, spent fuel samples (pellets) and samples doped with individual fission products have been inserted within a fresh fuel system to obtain reactivity worths [16,28]. Sufficient system perturbation to enable an accurate measure of reactivity worth typically requires isotope concentrations much greater than those present in a spent fuel sample. Unless the sample is large enough to provide a significant perturbation to the reference fresh fuel system, the reactivity worth cannot be easily calculated with conventional Monte Carlo codes that are typically used for criticality safety analyses. These experiments can thus provide a means to obtain validation of the reference cross sections used in the criticality analysis, but may be limited for use in typical validation approaches used by storage and transport cask vendors in the United States. The French program for burnup credit relies heavily on the use of reactivity worth measurements in conjunction with chemical assay data to demonstrate that the predicted fission product worths are conservative for their specific design codes and that the uncertainty in the fission product cross sections is encompassed by the uncertainty in the prediction of the fission product inventory.

Reactivity worth measurements [16,28], using small individual fission product samples and oscillation techniques, are more sensitive to the fission product cross-section uncertainties than the fission product solution criticals discussed in the previous section. The oscillation-type reactivity worth measurements provide more detailed information on the uncertainties in the individual fission product cross-section via evaluation of the uncertainties in the measured worth of each individual fission product. Unfortunately, this type of data can not be readily utilized within the typical code-to-experiment comparison approach to validation historically used in the U.S. for criticality safety analysis. This situation should change if, and when, the S/U methods of Ref. 26 are made available within publicly available production codes.

The U.S. DOE is currently exploring the potential and the benefits of obtaining fission product samples that can be used to perform similar reactivity worth measurements in a critical experiments facility or a research reactor. A DOE Nuclear Energy Research Initiative (NERI) project is also funded to investigate performance of worth experiments in the facilities at Sandia National Laboratories [29]. Current activities are directed on obtaining safety approvals; ideas on the specific type of worth measurements have not been formulated.

Reactivity worth measurements using portions of spent fuel assemblies are being planned as part of the REBUS experimental program [17] and the U.S. Nuclear Regulatory Commission (NRC) is participating in the program. These reactivity worth experiments provide Δk worths that will hopefully be large enough for evaluation with the actual Monte Carlo codes used in the safety analysis. When doing reactivity worth measurements with spent fuel segments, the experiment must either have accompanying destructive assays performed or the fuel design and reactor operation needs to be sufficiently characterized such that an integral-type benchmark (isotopic prediction and reactivity worth prediction) can be performed to provide a combined validation of both the depletion and criticality methodology. Such chemical assay measurements are planned as part of the REBUS program.

5. SUBCRITICAL EXPERIMENTS

The ideal experimental method for assessing the ability of a model to predict the multiplication factor of a system would be to place spent fuel in a cask or cask-like configuration and perform critical experiments. Such experiments are extremely challenging because it is extremely difficult to make even low-burnup spent fuel go critical in a controlled manner without first adding some fresh fuel. This is particularly true under cask conditions where external absorbers (basket material) are present. Spent fuel critical experiments are also complicated by the fact that the fuel samples are highly radioactive, and not as easily manipulated as unirradiated fuel. The expense and complexity of a spent fuel critical is further exacerbated by the need to determine the spent fuel composition by chemical assay (very expensive due to the potentially large number

of measurements required) or perform predictive analysis validated against other chemical assay information. At this time, no critical experiment using commercial spent fuel in a cask configuration is known to have been performed, although they have been studied [30].

An alternative to spent fuel critical experiments are subcritical multiplication measurements using spent fuel. Calculations could then be performed to show the capability to match the predicted multiplication factor to the measured value. As with a spent fuel critical experiment, this validation process would require predictions of spent fuel contents prior to the criticality calculations, and would therefore be an integral approach for validation. However, the spectrum should be very similar to that seen in a cask environment and the use of subcritical methods should allow increased flexibility in measuring different configurations. Besides the practical difficulty of handling spent fuel, the performance of subcritical measurements using spent fuel are made difficult by the practical difficulties with such measurements in a strong radiation field and the need to interpret k_{eff} from the actual measured quantities [31]. The accuracy of subcritical measurements in providing a k_{eff} value for validation is not as good as that provided by a critical experiment, but the advantage of having an actual spent fuel measurement and its potential to validate SNF cross sections (actual measured quantities are very sensitive to cross-section errors) means that such an experiment should be explored if additional measured data are deemed necessary.

6. REACTOR CRITICAL CONFIGURATIONS

A broad database of critical experiments with partially burned and spent fuel exists in the form of critical configurations within an operating reactor environment. During a commercial reactor startup, a controlled approach to criticality is always taken as part of the startup testing. The conditions at the point of criticality are well defined. Startups at the beginning of a fuel cycle contain a mixture of fresh and burned fuel, and often burnable poisons are present; startups occurring late in a fuel cycle are based on a combination of burned and spent fuel, and burnable poisons have typically been depleted.

Like the spent fuel experiments described earlier, the calculational model of a reactor critical configuration will require the prediction of spent fuel inventory for each assembly. Given the size of a commercial reactor combined with the variation in operating conditions during a fuel cycle, the task of estimating spent fuel contents at the time of a startup critical can be rather formidable. However, several reactor critical models have been developed with codes and data typically used for transport and storage cask safety analyses. The results [32 to 33] demonstrate the abilities of such codes to closely predict criticality under reactor conditions.

The advantage, of using reactor criticals in some fashion as part of the validation process, is that they provide measured critical values for systems actually containing SNF. The worth of the spent fuel, with respect to the fresh fuel and the degree to which fission products and boron poison concentrations impact the k_{eff} value, are issues that need to be considered when selecting critical configurations for validation. Just as with all of the other types of experiments discussed to this point, the reactor critical configurations do not provide the same neutronic environment as found in a flooded transport or storage cask. For example, the presence of fresh or partially burned fuel in the reactor and the physical differences between a cask and reactor conditions (e.g., soluble boron versus boron plates) can cause changes in the governing spectrum. Thus, the S/U methodology is also being used to further explore the value of the reactor critical configurations for code and data validation. Three PWR commercial reactor critical state points have been analyzed using the S/U methodology and comparisons made with SNF cask environments. The results indicate that the reactor critical state points have adequate similarity to cask environments. Reactor critical configurations are the only measured information where significant quantities of SNF are used and, from an integral perspective, provide a viable source of validation information for both actinides and fission products.

7. SUMMARY AND DISCUSSION

The purpose of a validation activity is to assess the capability of the codes and data to predict reality. As used in criticality safety, the validation process should be able to demonstrate the bias and the uncertainties associated with the analysis code(s) and data. The overall uncertainties can arise from uncertainties in the experiments, uncertainties inherent in the code models and data, and uncertainties specific to the user (model approximations, selection of code options, etc.). Currently, approaches used in the United States for criticality safety validation often apply statistical techniques to derive "bounding" estimates of the bias and uncertainty based on the differences between critical ($k_{eff} = 1.0$) and the computed result, together with the spread in the computed results [24]. As demonstrated from the previous subsection, no one set of critical experiments can provide adequate validation for burnup credit applications using this approach alone. Thus, the necessary approach involves utilization of all applicable experimental information in a manner that reasonably ensures that bounding estimates have been determined for the bias and uncertainty. Work at ORNL is focused on quantifying the value of each type of experiment and investigating validation approaches that effectively combine analysis correlations with the types of experimental information.

The DOE Topical report [21] used only fresh fuel critical experiments to validate the analysis of k_{eff} for spent fuel casks and incorporated the nuclide bias and uncertainty by separate adjustment of the predicted SNF isotopics based on comparison with chemical assay data. The limited database of chemical assays and the difficulty and/or uncertainty inherent in the measurement of many of the individual nuclides (most fission products and minor actinides) pose significant obstacles to this phase of the validation process. Even with additional measurements, relatively few data points will ever be available relative to the variety of fuel designs and operating histories to be considered. Thus, given a database with such a limited sample size, it is difficult to obtain meaningful statistics relating predictions as a function of spent fuel characteristics (enrichment, burnup, fuel design, etc.). The critical experiments proposed in Ref. 21 for actinide-only validation include all the nuclides of the actinide-only inventory used in the safety assessment, although the concentrations and combinations are not that observed in spent fuel. To overcome the limitations of the fresh fuel critical experiments relative to their material compositions, use of reactor-critical configurations have been proposed [34] and studied [32,33].

Integral validation involves the use of depletion methods coupled with criticality calculations to determine k_{eff} for a measured system containing SNF (e.g., a spent fuel critical or reactor critical configurations). In practice, this procedure would be applied in spent fuel cask calculations. The perceived limitation with integral validation is that the biases and uncertainties for the depletion approach cannot be separated from those associated with the criticality calculation, and only the net biases and uncertainties in the entire procedure are obtained. Integral validation allows for compensating errors in the depletion approach (i.e., underprediction of a given nuclide's concentration coupled with simultaneous overprediction of a different nuclide's inventory). Thus, it is desirable to ensure the uncertainty estimated for individual nuclides is understood and properly considered in the safety analysis.

Arguments have been made that reactor critical configurations are not appropriate even for integral validation because of differences between reactor conditions and cask conditions. However, other arguments can be made in favor of reactor critical configurations as integral benchmarks, primarily because the design and material composition of the fuel to be placed in a cask is identical to that present in commercial reactors. Thus, the issue with reactor critical configurations is their relevance to cask geometries, whereas the issue with fresh fuel critical experiments is their relevance to the inventory of SNF. Both issues are appropriate when discussing the relevance of reactivity worth measurements.

Initial work with S/U methods [27] have indicated there is some benefit in utilizing each of the types of experimental data; the challenge will be appropriate combination of the data into an adequate validation process. The S/U methods have been used in a preliminary fashion to address the applicability of the fresh fuel critical experiments and reactor critical configurations to cask designs with SNF. Besides defining the

applicability of these classes of experiments, S/U analysis can potentially identify deficiencies in the current database and provide a quantitative basis for extension beyond the existing database. The S/U methods also have the potential to assess the benefit of reactivity worth experiments and subcritical experiments and identify potential reductions in the margin of subcriticality allowed by their use. Reactivity worth experiments and subcritical experiments are prime candidates to support additional validation of fission product cross sections, which have typically not had the scrutiny or intense evaluation dedicated to the primary actinides.

Sensitivity/uncertainty methods may also be an approach that can be used to support expansion of the area of applicability for the chemical assays beyond their current limits in terms of burnup and initial enrichment (see Fig. 1). In other words, does 3.6 wt % fuel burned to 40 GWd/t have similar irradiation characteristics as 4.5 wt % fuel burned to 55 GWd/t? If sensitivity methods can be used to quantify the similarity between different SNF characteristics, then S/U methods may be able to establish and justify trends such that interpolation and/or extrapolation techniques can be used to estimate the bias and uncertainty associated with SNF for which there is no chemical assay data.

For actinide-only burnup credit in transport and storage cask applications, it appears there is sufficient experimental data available to enable reasonable validation of the codes and data — although the best process for combining and using the various data may be debated and the bias and uncertainties may not be as low as desired. However, for burnup credit cask applications with fission products, there is a significant paucity of readily available measured data and the validation process acceptable for using the measured data is still in its infancy. A prudent approach to burnup credit validation should involve assay data validation, followed by cross-section validation for the actinides and fission products. The existing mixed-oxide fuel criticals combined with French HTC experiments [25] are believed to be sufficient for actinide-only cross-section validation purposes. Additionally, applications that take credit for fission products need to consider experiments that validate individual fission-product cross sections. Validation may be best accomplished by a combined approach of large-sample, individual fission product worth measurements, such as the French fission product solution criticals [26] or the DOE/NERI experiments [27], and the small-sample, individual doped-fission product worth measurements like those of Refs. 16 and 28. Although more complex to model, commercial reactor critical data provide a valuable source of experimental information for integral validation of the SNF compositions and cross sections and the effect of neutronic interaction between assemblies. Work needs to continue to best determine how to incorporate this information into the validation process for use in transport and storage cask licensing.

Utilizing the negative reactivity credit from fission products continues to be a goal for optimum use of burnup credit in the United States. One approach that has been offered [35] is to quantify two independent factors to account for the effects of isotopic prediction inaccuracies and isotope cross-section inaccuracies (derived from reactivity worth measurements). The product of these two factors and the predicted worth values in the cask configuration gives an estimate of the "guaranteed" fission-product worth in the cask application of interest. Efforts are underway to quantify these effects for an example application and investigate the methodology for use in transport and storage applications. Similarly the U.S. DOE is seeking burnup credit that includes fission product credit in their efforts to license the permanent SNF repository [36]. The DOE efforts rely heavily on the use of reactor critical configurations with fresh fuel criticals and chemical assay data being used as appropriate.

ACKNOWLEDGEMENTS

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