Full Burnup Credit in Transport and Storage Casks: Benefits and Implementation

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Abstract – *The benefits of burnup credit and the technical issues associated with utilizing burnup credit in spent nuclear fuel (SNF) transportation casks have been studied in the United States for almost two decades. The issuance of the U.S. Nuclear Regulatory Commission (NRC) staff guidance for actinide-only burnup credit in 2002 was a significant step toward providing a regulatory framework for using burnup credit in transport and storage casks. However, adherence to the current regulatory guidance (e.g., limit credit to actinides) enables only about 30% of the existing pressurized water-reactor (PWR) SNF inventory to be transported in high-capacity (e.g., 32-assembly) casks. Work has been performed to demonstrate that the allowable inventory percentage could increase to nearly 90% if credit for actinides and fission products were allowed (i.e., "full" burnup credit). This paper discusses the benefits of full burnup credit and presents a technical strategy designed to obtain and evaluate experimental data to support a strong safety basis for fission product credit. This paper is adapted from a paper presented by the authors at the 2006 International High-Level Radioactive Waste Management Conference.*

I. INTRODUCTION

Safe, efficient, and effective management of spent nuclear fuel (SNF) from commercial nuclear power plants will require increasing attention to transport and, potentially, dry storage in casks. Historically, cask designs for transporting SNF have had to demonstrate criticality safety and structural integrity while meeting limits on weight, thermal loading, external dose, and containment. With the reduced thermal load and dose provided by a minimum 5-year cooling time for transport of SNF, it became apparent in the late 1980s that SNF cask capacity would often be limited by the conservative, yet simple assumption of unirradiated fuel (i.e., no credit for the fuel burnup) used in criticality safety evaluations. For pressurized-water-reactor (PWR) SNF, burnup credit eliminates the need for the gapped basket structures (i.e., flux traps) used for separation and criticality control—thus providing an important degree of flexibility to cask designers. Elimination of the flux traps can provide additional space that allows high-density packing of the SNF for capacity increases of at least 30% in PWR casks.

Although crediting the reactivity reduction from burnup (i.e., burnup credit) is an important component of enabling SNF casks to have high capacity, the current regulatory guidance recommends credit only for the reactivity change due to major actinides. The current regulatory position for transport and storage is provided in the U.S. Nuclear Regulatory Commission (NRC) Interim Staff Guidance 8, Revision 2 (ISG-8R2) which was issued by the Spent Fuel Project Office in 2002. Adherence to this guidance will enable no more than ~30% of the domestic SNF inventory from PWRs to be loaded in high-capacity (~32-PWR-assembly) casks that are planned for transport. However, additional burnup credit provided by fission products will enable high-capacity casks to handle the bulk (up to 90%) of the domestic PWR SNF inventory.

In 2004, Oak Ridge National Laboratory (ORNL) prepared a roadmap for a project whose goal was to develop and/or obtain the scientific and technical information necessary to support preparation and review of a safety evaluation for transportation cask designs that use full (actinide and fission product) burnup credit for PWR contents. Subsequently ORNL worked cooperatively with the NRC, the Electric Power Research Institute (EPRI), and the U.S. Department of Energy (DOE) Office of Logistics Management to execute the project plan. The plan called for existing critical experiments and assay measurement data to be assessed for technical value in developing an adequate safety evaluation that includes both actinide and fission product credit. New data would be acquired based on the needs identified following assessment of existing data.

The decision by the DOE Office of Civilian Radioactive Waste Management (OCRWM) to specify use of a relatively low capacity (e.g., 21 PWR assemblies) canister for Transportation, Aging, and Disposal (TAD) means that criticality control for the TAD canister can likely be achieved without full burnup credit. The cost benefits discussed in this paper were performed prior to the decision on the TAD canister system and are applicable for a scenario where the goal is to minimize shipments through use of high-capacity casks. However, the technical information discussed in this paper may still be needed to transport SNF already loaded in high-capacity storage casks and may be beneficial for transport of a portion of the SNF inventory that may not be transported via the TAD. In addition, the technical strategies discussed in this paper are also now being pursued to help facilitate the safety basis for permanent disposal, where criticality control by flux traps cannot always be assured due to the potential environment and events that must be considered.

II. ASSESSMENT OF BENEFITS FOR FULL BURNUP CREDIT

II.A. Inventory Accommodation for PWR SNF

During 2004, the DOE Energy Information Administration released a Microsoft Access™ database with an updated version of the RW-859 compilation submitted by U.S. commercial nuclear power plant licensees for PWR SNF through the end of 2002. In the present study, 6 PWR fuel assembly types that comprise about 94% of the 70,290 PWR SNF assemblies in the database were investigated to assess the benefits that would be provided by full burnup credit assuming transport in a high-capacity cask.

A generic high-capacity (32-assembly) cask, designated GBC-32, was selected as the reference configuration to assess the benefits of full burnup credit for the RW-859 inventory. The GBC-32 cask is representative of burnup-credit rail casks currently being considered by U.S. industry and, prior to the issuance of the TAD specification, was judged a relevant and appropriate configuration for this evaluation. The loading curves (required burnup vs initial enrichment) were generated with basic assumptions (reactor operating conditions, bias and uncertainty process, axial profiles, etc.) consistent with ISG-8R2.

The acceptability of the SNF assemblies for the six fuel types is summarized in Table I. Consistent with the regulatory guidance, assemblies that require burnup >50 GWd/MTU are classified as unacceptable. Also, the determination of acceptability does not account for burnup uncertainty, which would reduce the percentage of acceptable assemblies. The results indicate that while burnup credit with ISG-8R2 can enable loading a large percentage of the Combustion Engineering (CE) 14 × 14 and Westinghouse Electric (WE) 14 × 14 assemblies in a high-capacity cask, the benefits of ISG-8R2 are minimal for the other assembly designs considered.

To evaluate the effect of selected calculational assumptions, Fig. 1 compares the reference case loading curve for the WE 17 × 17 assembly with loading curves for the following individual variations:

1. Inclusion of minor actinides (^{236}U, ^{237}Np, ^{243}Am) and five of the six principal fission products (^{149}Sm, ^{143}Nd, ^{151}Sm, ^{133}Cs, and ^{155}Gd), with isotopic correction factors based on comparisons with available assay data.
2. Inclusion of minor actinides and five principal fission products with spent fuel composition bias and uncertainty based on a best-estimate approach for bounding isotopic validation.
3. Inclusion of sixteen fission products (^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{143}Nd, ^{145}Nd, ^{151}Eu, ^{153}Eu, ^{155}Gd) and minor actinides (^{236}U, ^{237}Np, ^{243}Am), with spent fuel composition bias and uncertainty based on a best-estimate approach for bounding isotopic validation.
4. Inclusion of the principal fission products and minor actinides without any correction for isotopic validation.

Note that for a few of the relevant fission products (e.g., ^{103}Rh), insufficient measured assay data are available to estimate bias and uncertainty. Thus, with the exception of the final case, no credit was taken for their presence in the SNF.

All of the curves in Fig. 1 were prepared assuming a 5-year cooling time. Extending the cooling time up to 20 years makes only a marginal increase in the allowed inventory. A more-effective approach is shown in Fig. 1, where inclusion of fission products and/or the use of more-realistic approaches to isotopic validation offer significantly larger increases in allowed inventory. For the GBC-32 cask, the percentage of acceptable assemblies increases from 9 to 38% with the inclusion of the five primary fission products and minor actinides (both cases at 5-year cooling), and from 38 to 78% with the use of a bounding best-estimate approach for isotopic validation. The next case includes the remainder of the principal fission products and uses the best-estimate isotopic validation approach. These assumptions allow the percentage of acceptable assemblies to increase to 90%. The final case shown in Fig. 1 corresponds to full credit for the calculated actinide and principal fission product compositions and, given the conditions considered, represents a limit in terms of the available negative reactivity that might potentially be credited.

TABLE I. Summary of SNF acceptability in the GBC-32 cask with actinide-only burnup credit for the six assembly types considered

Assembly type	Total in discharge data	Number acceptable for loading	Number unacceptable for loading
CE 14 × 14	6,972	4,518 (65%)	2,454 (35%)
CE 16 × 16	6,828	1,731 (25%)	5,097 (75%)
B&W 15 × 15	7,519	166 (2%)	7,353 (98%)
WE 17 × 17	28,704	2,448 (9%)	26,256 (91%)
WE 15 × 15	10,365	475 (5%)	9,890 (95%)
WE 14 × 14	5,448	4,686 (86%)	762 (14%)
Total	65,836	14,024 (21%)	51,812 (79%)

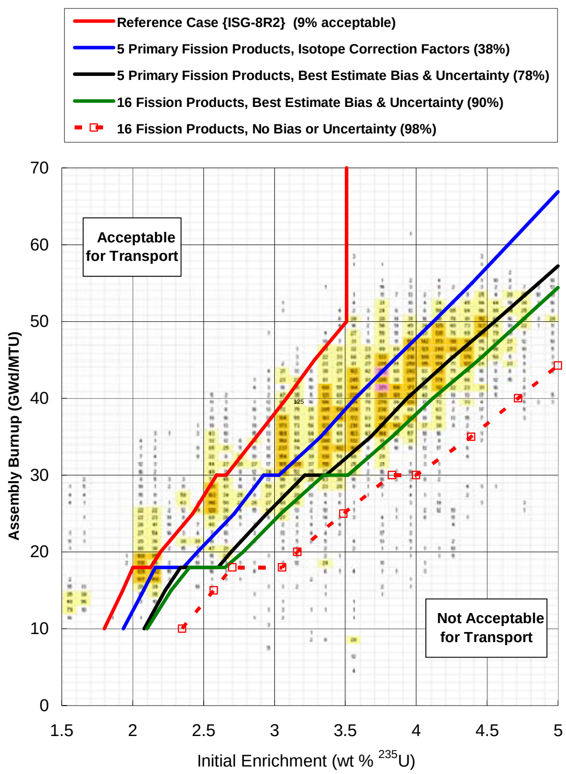

Fig. 1. Comparison of calculational assumptions for WE 17 × 17 fuel assemblies. Percentages of inventory acceptable for the GBC-32 cask are shown in parentheses.

Comparison of actinide-only-based loading curves for the GBC-32 cask with PWR SNF discharge data (through the end of 2002) leads to the conclusion that additional negative reactivity is necessary to accommodate the majority of PWR SNF assemblies in high-capacity casks. Relatively small shifts in a cask loading curve, which increase or decrease the minimum required burnup for a given enrichment, can have a significant impact on the number of SNF assemblies that are acceptable for loading. Thus, as the uncertainties and corresponding conservatisms in burnup credit analyses are better understood and reduced through acquisition of additional experimental data, the population of SNF acceptable for loading in high-capacity casks will increase. Figure 1 demonstrates that given appropriate data for validation, the most significant component that would improve accuracy, and subsequently enhance the utilization of burnup credit, is the inclusion of fission products. This fact is valid for any high-density SNF system design.

II.B. Pre-TAD Cost-Benefit Assessment for PWR SNF

Assuming a scenario where shipments to a permanent repository are performed using high-capacity casks, a relatively simple evaluation of the potential cost benefits for using full burnup credit was performed in 2004. This evaluation used the current capacity limit for the Yucca Mountain repository [70,000 metric tons of heavy metal (MTHM)], the percentage of total MTHM from PWRs at the end of 1998 (~64%), and the average number of PWR assemblies per MTHM to predict that ~100,000 PWR assemblies will need to be transported to the repository. Using representative loading curves and assuming that assemblies that cannot be accommodated in 32-assembly casks are transported in 24-assembly casks, it was estimated that full burnup credit can reduce the number of shipments by ~22% (~940 shipments) while actinide-only-based burnup credit reduces the number of shipments by only ~8% (~315 shipments). An ad hoc survey of industry experts suggested an estimated cost per rail cask shipment (freight and operational costs) ranging from \$200K to \$500K. Although the majority of the experts supported the \$500K/shipment value, a conservative estimate of \$250K was adopted. Using this per-shipment estimate [assuming shipments are reduced by 625 = (940 – 315)] provides a resulting costs savings of at least \$156M that can be realized from establishing full burnup credit for SNF transportation.

A significant simplifying assumption used in the above cost evaluation is that all assemblies would be loaded and transported in large (100- to 125-ton) rail-type casks. The initial cost estimate was updated in 2005 to remove this simplifying assumption and investigate the impact of using a cask fleet of varying sizes. Discharge data as a function of site cask capabilities (size of cask that could be handled) were obtained and estimates were developed for (1) cost per cask shipment (varying from $150K for truck cask to $250K for 32-element rail cask); (2) cask design capacities with and without burnup credit (varying from 100% for a legal-weight truck cask to 30% for large rail casks); and (3) percentages of assemblies acceptable for loading with and without burnup credit (based on approximate loading curves for each cask using actinide-only and full burnup credit). Using this information, the cost savings associated with burnup credit for transportation are estimated to be ~$638M. Of this total, ~$235M is attributable to credit for fission products. The cost estimates are higher than the simpler cost-benefit analysis provided above because there is an increased shipment cost on a per-assembly basis associated with the use of smaller casks. Thus, even if the mix of casks as assumed is not correct, the $156M figure based solely on a rail cask appears to be a minimal savings assuming that other cask design constraints (e.g., decay heat) do not limit the burnup credit benefits. These two cost-benefit estimates demonstrate the significant potential cost savings associated with establishing full burnup credit for the specified scenario (i.e., shipments minimized through the use of high-capacity casks).

III. DATABASE OF CRITICAL EXPERIMENTS FOR FULL BURNUP CREDIT

III.A. Background and Approach

To achieve the potential benefits discussed and demonstrated in Section II, ORNL developed and initiated a plan to obtain the data needed for straightforward and effective preparation and review of a criticality safety evaluation with full burnup credit. NRC staff have noted that the rationale for restricting ISG-8R2 to actinide-only is based largely on the lack of clear, definitive experiments that can be used to estimate the bias and uncertainty for computational analyses associated with using burnup credit. Applicants and regulatory reviewers are constrained by both a scarcity of data and a lack of clear technical bases (e.g., criteria) for demonstrating applicability of the data.

The goal therefore is to obtain, and make available to industry, a well-qualified experimental database that can ensure reliable and accurate estimation of any bias and uncertainty resulting from the codes and data used to predict the system neutron multiplication factor, k_{eff}. Rather than an a priori decision on suitability of candidate experiments, ORNL sought to obtain and assess critical experiment data from the following sources:

1. critical experiments within the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (*IHECSBE*);
2. proprietary critical experiment data;
3. commercial reactor criticals (CRCs), i.e., critical state points from operating reactors; and
4. proposed new critical experiments.

The applicability and value of this database of critical experiments is currently being assessed with the aid of sensitivity and uncertainty (S/U) analysis tools developed at ORNL and incorporated within Version 5 of the SCALE code system. The TSUNAMI-3D sequence within SCALE uses first-order linear perturbation theory to calculate the sensitivity of k_{eff} for systems (e.g., SNF casks) and/or critical experiments to variations in nuclear data. Energy-, nuclide-, reaction-, and position-dependent sensitivity profiles are generated and saved in sensitivity data files. The TSUNAMI-IP module of SCALE uses the sensitivity data file information and cross-section uncertainty data to evaluate the similarity of different systems. One of the products of this comparison is an integral index, referred to as c_k, which is a single-valued quantity used to assess the similarity of uncertainty-weighted sensitivity profiles for all nuclide reactions between a modeled system and a critical experiment. A c_k index is similar to a correlation coefficient, and a value of 1 indicates that the compared systems have identical uncertainty-weighted sensitivities. A value of 0 indicates that the systems are completely dissimilar. The current guidance is that critical experiments with a c_k value of at least 0.9 are applicable for validation purposes and that c_k values between 0.8 and 0.9 indicate marginal applicability.

The SCALE S/U tools were used to analyze the GBC-32 prototypical high-capacity rail cask loaded with WE 17 × 17 fuel having accumulated burnups of 10 to 60 GWd/MTU. The results from this cask model serve as the reference for applicability comparisons with the sets of critical experiments under consideration.

III.B. Assessment of IHECSBE and French Proprietary Experiments

As part of this project, ORNL was able to negotiate a multi-option contract with Cogema to gain access to proprietary critical experiments performed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) at their Valduc critical experiment facility. These experiments are part of a larger French program to develop a technical basis for burnup credit. Subsequent to a positive evaluation and acquisition rights, the data obtained will be made available to industry for use in cask design and licensing activities.

ORNL has received the first set of critical experiment data documented using the format of the IHECSBE. These experiments were performed with rods having uranium and plutonium isotopic compositions similar to $U(4.5\%)O_2$ fuel with a burnup of 37,500 MWd/MTU. The experimental series, referred to as the HTC experiments, investigated 156 configurations divided into 4 groups, as illustrated in Fig. 2. The first group is a single clean-water-moderated and water-reflected array of HTC rods with the pin pitch varied from 1.3 to 2.3 cm. The second group is similar to the first, except that boron or gadolinium is dissolved in the water in varying concentrations. The third group has 4 separate assemblies of HTC rods, separated by varying distances, and with borated steel, Boral™, or cadmium plates on the outsides of the assemblies in 11 of the critical configurations. The fourth group is similar to the third group, except that a thick lead or steel shield is placed around the outside of the four assemblies to simulate the type of reflector representative of a cask.

These 156 HTC critical experiments, together with nearly 1000 critical configurations from the IHECSBE, have been analyzed with the TSUNAMI-IP sequence, and the sensitivity data obtained have been compared with sensitivity data for the reference cask model loaded with assemblies burned to 40 GWd/MTU. Figure 3 shows the distribution of the c_k values for the 1134 critical configurations when compared with the reference burnup credit cask model. As shown in the figure, the 170 ^{233}U experiments, the 150 high-enrichment-uranium experiments, the 4 intermediate-enrichment-uranium experiments, the 197 plutonium-only configurations, and the 256 low-enrichment-uranium experiments all have c_k values of <0.8—an expected result given the difference in fissile material between these critical experiments and SNF. Only 45 of the 201 non-HTC mixed-oxide (MOX) configurations have c_k values ≥ 0.8, with none having c_k values ≥ 0.9. Additional non-HTC MOX experiments continue to be assessed. However, the strong applicability of the HTC MOX experiments is demonstrated by the fact that 152 of the 156 configurations have c_k values ≥ 0.8, with 143 c_k values ≥ 0.9. The results of these studies confirm the significant value of the HTC experiments for criticality validation of the primary actinides and the weaker validation basis that exists without the HTC experiments.

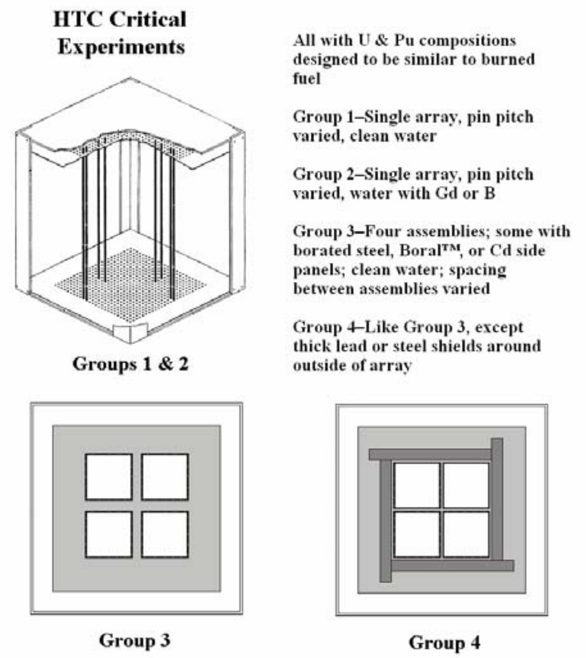

Fig. 2. French HTC critical experiments.

Work has been initiated to assess two sets of critical experiments for validating the fission product component of SNF in a cask environment. The first set of experiments was performed in 2003 at Sandia National Laboratories (SNL) as part of a DOE Nuclear Energy Research Initiative (NERI). The set of experiments included thin ^{103}Rh foils stacked between fuel pellets in UO_2 rods placed in a hexagonal array. S/U analyses have been performed for the SNL ^{103}Rh critical experiments, and the results have been compared with S/U analyses results for the GBC-32 cask model. A comparison of the energy-dependent sensitivity profiles shows reasonably good agreement except in the 1- to 2-eV neutron energy range. Studies have been performed to show how a modified experiment design (use of thinner foils) could improve the applicability of the experiments. The S/U tools will be employed in the design process of planned SNL experiments to ensure maximum applicability.

The second series of experiments being assessed for their value in validation of the fission product burnup credit are the second set of critical experiments performed by IRSN and included as part of the contract ORNL currently has with Cogema. ORNL has received preliminary reports that describe 147 critical configurations (referred to as the "PF" experiments), 74 of which contain fission products. ORNL will work to perform S/U analyses for these French fission product experiments using TSUNAMI-3D and TSUNAMI-IP. Upon completion of an evaluation that ensures these PF experiments can provide a reliable estimate of the bias and uncertainty for fission product burnup credit, rights to use the data will be acquired and the data distributed for use by potential licensees.

III.C. Assessment of Commercial Reactor Critical Configurations

Reactor core configurations and material compositions for 33 Crystal River Unit 3 state points were obtained from the Yucca Mountain Project (YMP) and S/U analyses have been performed to investigate the applicability of the CRC state points to burnup credit validation. The CRC state points require very large complex computational models with the following information required for accurate modeling: fuel assembly locations during reactor cycles and 18-node fuel rod compositions; burnable poison rod assembly (BPRA) core locations and 17-node compositions; rod cluster control assembly (RCCA) and axial power shaping rod assembly (APSRA) core locations, compositions, and insertion heights; and a description of assembly hardware.

Preliminary results for the Crystal River CRC state points show c_k >0.90 for 25 of the 33 cases with effective full power days ranging from 0 to 515. In addition, comparisons of the sensitivity files show reasonable similarity for many of the key fission products. However, a major drawback to use of the CRC state points is that the lack of specific information (exact component locations, spatially varying operating conditions, and isotopic compositions, etc.) for these measured critical systems leads to major difficulties in quantifying the "experiment" uncertainties such that the CRCs can be effectively used to establish the bias and uncertainty for the computational tools.

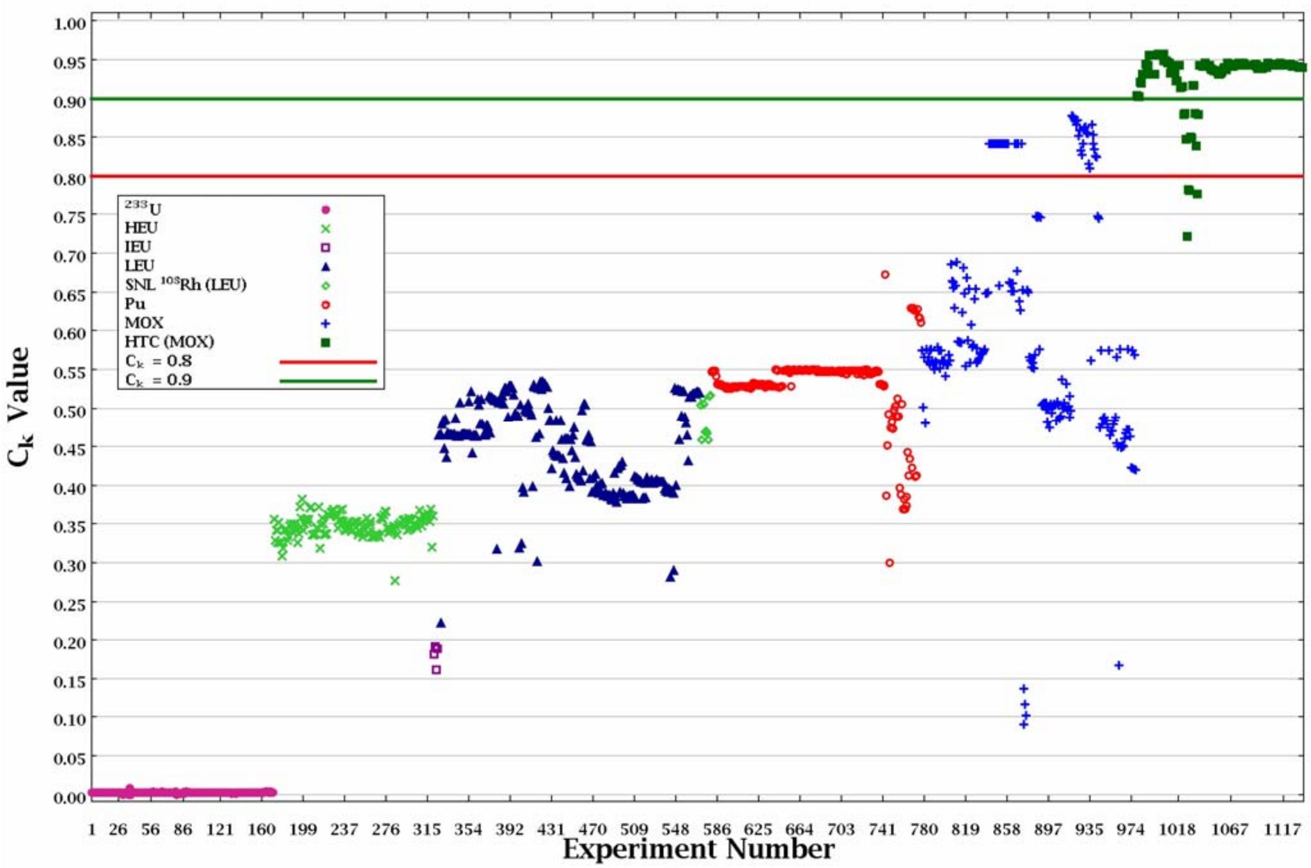

Fig 3. Critical experiment applicability to burnup credit, showing 1134 critical experiments compared with PWR model

III.D. Proposed New Critical Experiments

In parallel with efforts to locate and evaluate existing critical experiment data, ORNL has also worked with SNL to pursue planning activities for performing additional experiments with the principal fission products. The planned experiments would be performed at SNL in a manner similar to the critical experiments with ^{103}Rh performed under the DOE/NERI project. The S/U analysis tools, which were not available when the ^{103}Rh critical experiments were designed, will be used to help guide the design of the critical configurations. The goal will be to address any technical needs that may not be adequately addressed with the data obtained from the French critical experiments. Initial planning activities are under way and, given sponsor support, critical experiments are expected to begin in 2008.

IV. DATABASE OF ISOTOPIC ASSAY DATA FOR PWR FULL BURNUP CREDIT

IV.A. Evaluated Assay Data for Fission Products

Just as there are limited benchmark critical experiments that can be used to estimate the bias and uncertainty due to the presence of fission products in SNF cask systems, the existing regulatory guidance of ISG-8R2 notes a definitive lack of measurements that can be applied to estimate the bias and uncertainty in the prediction of the fission product compositions in SNF.

Regardless of the burnup, the top six nongaseous fission products, accounting for approximately 75% of the total worth of all fission products, are ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{151}Sm, and ^{155}Gd. These six fission products have been the focus of ORNL's efforts to obtain and assess potential sources of measured data that can support a strengthened technical basis for fission product credit.

Although radiochemical assay measurements have been reported for a large number of spent fuel samples, most measurements include only the major actinides. Relatively few measurements include the largely stable fission products important to burnup credit (i.e., ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{151}Sm, ^{152}Sm, ^{155}Gd, and ^{153}Eu). Of the 56 PWR spent fuel samples that had been evaluated by ORNL prior to 2005, only 19 included measurements for any of these fission products, and many samples have measurements for only a small number of fission products. No measurements are available for three fission products (^{95}Mo, ^{101}Ru, and ^{109}Ag), and ^{103}Rh had just one measurement. Table II provides a summary of the total number of measurements assessed and used by ORNL for each fission product in general order of descending importance. The fission product assay measurements shown in Table II are from just two reactors: the Calvert Cliffs fuels [designated as Approved Testing Materials (ATM)-103, ATM-104, and ATM-106 fuels] measured by Pacific Northwest National Laboratory (PNNL) and the V. G. Khlopin Radium Institute (St. Petersburg, Russia) and the Japanese Takahama Unit 3 PWR fuel measurements performed by the Japan Atomic Energy Research Institute.

In 2005, ORNL performed a thorough review of existing information on measured assay data with the goals of (1) collecting all of the relevant data into a single database and (2) identifying measurement data that are not currently being utilized. The calculated-to-experiment (C/E) ratios obtained for the measurements noted in Table II were used to investigate the potential improvement (additional negative reactivity that could be credited) that would be obtained with availability of similar quality measurements. At least 20 high-quality measured samples need to be available to provide a good statistical estimate of the isotopic uncertainty, and avoid significant statistical penalties due to low sample size. Thus, the goal is to have this minimum number of measurements available for the validation of the principal fission product nuclides. The samples must also cover the range of spent fuel characteristics and variations. Additional samples may be required to evaluate trends identified in the isotopic bias.

IV.B. Sources of Additional Assay Data—Proprietary

This section describes potential foreign sources of isotopic assay data that ORNL has explored as a means to support code validation for burnup credit using fission products. The sources include existing proprietary programs, currently active programs, and opportunities to perform new measurements.

The Commissariat à l'Énergie Atomique (CEA) of France has established experimental programs to provide data for the validation of French computer codes. The programs include spent fuel assay measurements in support of fuel inventory and fuel cycle studies, including burnup credit. The data from these programs are proprietary. However, through the contract with Cogema (one of the optional purchases under the contract discussed in Section III), ORNL can obtain and distribute the data for use with burnup credit design and review activities. The available Bugey reactor assay measurements include only two SNF samples of 2.1 wt % and 3.1 wt % enrichment, with burnup less than 38 GWd/MTU. The available Gravelines reactor assay measurements include three SNF samples with an initial enrichments of 4.5 wt % and burnup values of 39.1, 51.6, and 61.2 GWd/MTU. All of these samples include measurements for the fission products of interest. If the CEA data are acquired, assay measurements for three boiling-water reactor (BWR) SNF samples from the German Gundremmingen reactor would also be provided.

The CEA fission product data are viewed as highly beneficial to strengthening the technical basis to support quantifying fission product uncertainty because of (1) the high-precision radiochemical analysis methods employed, (2) the range of enrichments and burnups (covering most commercial U.S. fuels), (3) the use of standard commercial fuel assemblies (non-rebuilt), and (4) the fact that the fuel is probably well characterized (because it was selected specifically to support code validation in France). However, the quantity of CEA fission product assay data is limited to 5 PWR samples, thus leaving the total number of measurements available for many nuclides well below the target value of 15 to 20.

TABLE II. Number of PWR assay measurements and relative importance of fission products to burnup credit

(Higher Importance)														(Lower Importance)
^{149}Sm	^{143}Nd	^{103}Rh	^{151}Sm	^{133}Cs	^{155}Gd	^{152}Sm	^{99}Tc	^{145}Nd	^{153}Eu	^{147}Sm	^{109}Ag	^{95}Mo	^{150}Sm	^{101}Ru
9	14	1	9	3	4	9	9	14	4	9	0	0	9	0

Belgonucleaire is coordinating the international REBUS program to obtain worth measurements for SNF and the MALIBU program to obtain isotopic assay data for high-burnup spent fuel. Similarly, there has been a research program of the Spanish organizations CSN, ENUSA, and ENRESA to obtain comprehensive isotopic characterization on high-burnup PWR fuel. Through support from NRC and DOE, ORNL is participating in REBUS and MALIBU, and collaborating with ENUSA to obtain high-quality fission product assay data. The REBUS program will provide fission product assay data for one PWR UO_2 SNF sample, the MALIBU program will provide fission product assay data for two PWR UO_2 SNF samples, and the Spanish program will provide fission product assay data for seven PWR SNF samples. These data will be evaluated by ORNL and distributed as needed at the end of the proprietary period established by each program.

IV.C. Sources of Additional Assay Data—Nonproprietary

In 2005, ORNL contracted with the PNNL to investigate and assess whether there are existing U.S.-origin SNF samples that can be retrieved and made available for expanding the database of radiochemical assay data for validation of fission product burnup credit. A large percentage of the existing U.S. fission product assay data was generated by the Material Characterization Center (MCC) at PNNL as part of a DOE/RW program conducted in the late 1980s and early 1990s. PNNL's investigation identified numerous well-characterized SNF samples that have the criteria (e.g., enrichment, burnup, and fuel type) needed to address shortages in the available assay database. Subject to sponsor funding, it is anticipated that radiochemical assay measurements will begin in 2007.

A major activity at ORNL has been a reassessment of reported measurements of Three Mile Island Unit 1 (TMI-1) SNF that were performed circa 1999 to support the YMP. The TMI-1 measurements include 19 fuel samples with extensive fission product data. However, an earlier assessment of the TMI-1 data by ORNL and staff at the YMP showed the C/E results to be highly discrepant compared with the results from the other samples analyzed by ORNL and those reported by the CEA and Belgonucleaire programs for SNF with similar characteristics. For example, Ref. 20 reports differences of 30–40% between measured and calculated predictions for ^{239}Pu. Reanalysis performed by ORNL using state-of-the-art multidimensional lattice physics codes (both SCALE and HELIOS) still shows discrepancies of 10–20% as compared with typical C/E differences of ±5% for ^{239}Pu for other available assays.

The TMI-1 fuel was originally selected for post-irradiation examination because it had experienced extreme crud buildup during irradiation and possible fuel cladding failure of the assembly. The operating conditions experienced by these fuel samples were not well known, and it was postulated that uncertainties in local fuel conditions might significantly impact the predictions and explain the large C/E discrepancies. Because the origin of the uncharacteristically large discrepancies was unknown, the TMI-1 data were not deemed to be suitable for use in obtaining isotopic bias and uncertainties.

The difficulty in obtaining the quantity and quality of measured assay data for fission product nuclides has led ORNL to reinvestigate these samples. Through the assistance of EPRI and AREVA, ORNL was able to obtain additional assembly design information including the location and composition of gadolinium rods and assembly pitch associated with the TMI-1 assay samples. With this additional information, reanalysis performed by ORNL indicates that the eight high-precision TMI-1 samples measured at the GE-Vallecitos Nuclear Center provide C/E values consistent with that anticipated based on the experience with a range of reactor types and radiochemical assay programs. However, the 11 other samples remain discrepant. Further review indicates that the measurement methods used for these particular samples involved relatively low-precision techniques that resulted in large experimental uncertainties, likely making these results of limited value for code validation purposes. Thus, once the new information on the TMI-1 samples are documented and released, there should be at least an additional eight PWR SNF samples having a desirable initial enrichment (4.65 wt %) and burnup values (23–30 GWd/MTU) and providing high-quality measurements for many fission product nuclides. Measurement for the metallic fission products (Mo Tc, Ru, Rh, Ag) were not available for these 8 samples.

V. NUCLEAR DATA ASSESSMENT, MEASUREMENT, AND EVALUATION

The technical rigor (physics measurements and evaluations to smoothly fit data over the entire energy range) utilized in acquiring current fission product cross-section data is deficient relative to that for major actinides and can impact the uncertainty and credibility of the validation process. This discrepancy in technical rigor has long been a concern (albeit, a secondary concern, if sufficient integral assay and critical measurements with fission products are available) of NRC staff in their consideration of allowing fission product credit. Similar interests exist in Europe, and ORNL is working with the Institute for Reference Materials and Measurement under a DOE-Euratom agreement to assess the quality of cross-section data (from domestic and international sources) for the key fission product nuclides (i.e., ^{103}Rh, ^{143}Nd, ^{149}Sm, ^{151}Sm, ^{133}Cs, and ^{155}Gd) and propose new measurements as justified. Work has already been initiated on new measurements and an updated evaluation for ^{103}Rh.

VI. OTHER ACTIVITIES

VI.A. Data for Improved Safety Analyses

The reactor analysis used to predict the SNF composition for the burnup credit safety evaluation should assume operating history parameters that are realistically bounding in terms of the impact on the k_{eff} value. In an effort to provide a basis for statistically meaningful bounding values, ORNL has initiated an effort to gather and organize operational parameter data using the CRC information documented by the YMP. Soluble boron concentrations, maximum fuel temperature, and minimum moderator densities were the initial parameters investigated. Investigation of the range of data values obtained and the mean standard deviations will hopefully provide a technical basis for bounding assumption values that should be used in the safety analysis. Given a sufficiently large database, it is anticipated that there should be a reduction in the conservatism in the values recommended by ORNL in earlier reports. The reduction should allow a larger fraction of spent PWR fuel to be considered as acceptable for loading.

VI.B. Burnup Credit for Boiling-Water-Reactor SNF

The potential data needs for effective utilization of burnup credit in BWR SNF systems are likely to be less than that for PWR SNF systems. Each BWR assembly is less reactive than a PWR assembly, and criticality control is more easily achieved without the use of full burnup credit. ORNL has performed analyses that confirm the need for relatively little burnup credit in a high-capacity BWR SNF rail transport cask. In addition, analyses were performed to determine to what extent current high-capacity rail casks, which have a maximum initial enrichment limit of ~4.0 wt %, would need to be derated (capacity reduced) to accommodate maximum enrichment (5.0 wt %) BWR assemblies without burnup credit. The analyses suggest that a reduction in capacity of a 68-assembly cask to 64 assemblies will enable loading of 5.0 wt % BWR assemblies without credit for fuel burnup. Although the complexity of the reactor operation may be greater for BWR SNF than for PWR SNF, it is anticipated that the minimal amount of burnup credit that may be needed with BWR SNF systems can be readily achieved. The key need for any additional data would likely be for radiochemical assay data, but insufficient work has been done to address the specifics relative to transport and storage.

VII. SUMMARY

A simple but straightforward approach for quantifying the benefits of PWR fission product burnup credit was developed and indicates a savings in transport cost alone in the range of \$150M–\$400M.

The highest-priority data for critical experiments have been obtained (with the HTC critical experiment data set in final form and the PF, or fission product, critical experiment data set in draft form) from a proprietary French program and are currently being evaluated for applicability to SNF transport and storage casks. The initial results indicate that the HTC data set will provide a strong technical foundation for the actinide portion of burnup credit and enable more flexibility in the criteria by which credit for fission products is considered.

Radiochemical assay data needed for estimating bias and uncertainties in predicted fission product nuclides continue to be a challenge. ORNL has investigated known sources of assay data and is working to ensure existing data are available for safety evaluations.

The technical strategy discussed in this paper has pursued a diverse path in order to help (1) provide flexibility in future safety analyses and (2) ensure that a solid technical basis consistent with cost and benefit is established. Thus, critical experiment data continue to be assessed for applicability to cask systems, efforts to improve the cross-section data for fission product nuclides have been initiated, and activities are ongoing to increase the database via domestic efforts (e.g., new critical experiments at SNL and assay data measurements at PNNL) or international activities (e.g., participation in international research programs). The results of these activities will hopefully provide NRC with the information needed to consider an update to ISG-8R2 that provides guidance for implementing fission product credit.