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FULL BURNUP CREDIT IN TRANSPORT AND STORAGE CASKS: BENEFITS AND IMPLEMENTATION

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Full Burnup Credit in Transport and Storage Casks:
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Abstract – *The benefits of burnup credit and the technical issues associated with utilizing burnup credit in 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 towards 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) spent nuclear fuel (SNF) inventory to be transported in high-capacity (e.g., 32-assembly) casks. Work has been done 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 coordinated project that will: (1) obtain and evaluate experimental data to support a strong safety basis for fission product credit, (2) investigate unresolved technical issues associated with PWR full burnup credit, and (3) recommend approaches for boiling-water-reactor (BWR) burnup credit in transport and storage casks.*

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, 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 fuel 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 increases the capacity of PWR casks by at least 30%.

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 (a reduction in actinides that fission and an increase in actinides that absorb neutrons). The current regulatory position[1] for transport and storage is provided in the U.S. Nuclear Regulatory Commission's (NRC's) Interim Staff Guidance 8, Revision 2 (ISG-8R2). 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. Additional burnup credit provided by fission products is necessary to enable high-capacity casks to handle the bulk (up to 90%) of the domestic PWR SNF inventory.[2]

In 2004, Oak Ridge National Laboratory (ORNL) prepared a roadmap for a project whose goal is to develop and/or obtain the scientific and technical information necessary to support preparation and review of a safety evaluation for cask designs that use full (actinide and fission product) burnup credit to transport PWR SNF. Subsequently ORNL has worked cooperatively with the NRC, the Electric Power Research Institute (EPRI), and the U.S. Department of Energy (DOE) Office of National Transportation (ONT) to execute the project plan. Existing critical experiments and assay measurement data will be obtained and assessed for technical value in developing an adequate safety evaluation that includes both actinide and fission product credit. In addition, the benefits and potential use of burnup credit in boiling-water-reactor (BWR) SNF casks will be investigated.

II. ASSESSMENT OF BENEFITS FOR FULL BURNUP CREDIT

II.A. Inventory Accommodation for PWR SNF

During 2005, the DOE Energy Information Administration released a Microsoft Access™ data base with an updated version of the RW-859 compilation[3] submitted by U.S. commercial nuclear power plant licensees for PWR SNF through the end of 2002. The six PWR fuel assembly types—comprising about 94% of the 70,290 PWR SNF assemblies in the data base—were investigated to assess the benefits that would be provided by full burnup credit.

A generic high-capacity (32-assembly) cask, designated GBC-32, was selected as the reference configuration[4] 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 is therefore a relevant and appropriate configuration for this evaluation. The loading curves (required burnup vs initial enrichment) are generated with the STARBUCS sequence of the SCALE code system.[5] The basic assumptions (reactor operating conditions, bias and uncertainty process, axial profiles, etc.) can be found in Ref. 2.

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 CE 14 × 14 and WE 14 × 14 assemblies in a high-capacity cask, the benefits of ISG-8R2 is minimal for the other assembly designs considered.

To evaluate the effect of selected calculational assumptions, Figure 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 [6] 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 the principal 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 Figure 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 Figure 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.[6] 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 Figure 1 corresponds to full credit for the calculated actinide and principal fission product compositions and, given the conditions considered, represents an unattainable limit in terms of the potentially available negative reactivity.

TABLE I. Summary of SNF acceptability in the GBC-32 cask with actinide-only burnup credit for the four 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%)

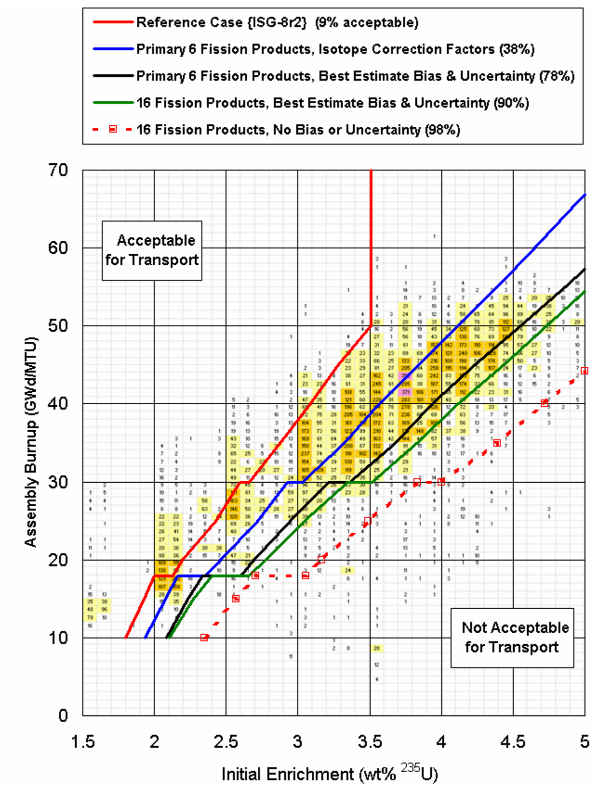

Figure 1. Comparison of calculational assumptions for WE 17 × 17 fuel assemblies. Percentages of inventory acceptable for the GBC-32 cask are shown in parentheses.

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, the population of SNF acceptable for loading in high-capacity casks will increase. 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.

II.B. Cost Benefits for PWR SNF

A relatively simple cost analysis of the potential benefits of burnup credit was initially performed using 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). A 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 analysis is that all assemblies would be loaded and transported in large (100 to 125-ton) rail-type casks. The cost estimate was updated 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. These estimates are consistent with the simpler analysis and demonstrate the significant potential cost savings associated with establishing burnup credit that includes credit for the fission product compositions. The cost estimates are larger 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. So 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 constraints (e.g., decay heat) do not limit the burnup credit benefits.

III. DATA BASE OF CRITICAL EXPERIMENTS FOR FULL BURNUP CREDIT

III.A. Background and Approach

To achieve the potential benefits discussed and demonstrated in Section II, ORNL has sought to obtain the data needed to enable straightforward and effective preparation and review of a criticality safety evaluation with full burnup credit. 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 associated with best-estimate analyses needed to obtain full 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.

Under this project, ORNL is working to obtain, and make available to industry, a well-qualified experimental data base 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 is seeking 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*);[7]
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 data base of critical experiments are being assessed using sensitivity and uncertainty (S/U) analysis tools developed at ORNL and incorporated within Version 5 of the SCALE code system.[5,8] The TSUNAMI-3D sequence within SCALE uses first-order linear perturbation theory[9] 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. TSUNAMI-IP 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, that is a single-valued quantity used to assess the similarity of uncertainty-weighted sensitivity profiles between a modeled system and a criticality experiment for all nuclide reactions. 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[8] 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[4] loaded with Westinghouse 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 at the Valduc research facility in France. These experiments are part of a larger French program[10] to develop a technical basis for burnup credit. Subsequent to assessment and evaluation, data obtained by ORNL 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 four groups, as illustrated in Figure 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 are dissolved in the water in varying concentrations. The third group has four 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 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. Only 45 of the 201 non-HTC mixed-oxide (MOX) configurations have c_k values $\geq$0.8, with none having c_k values $\geq$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 $\geq$0.8, with 143 c_k values $\geq$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.

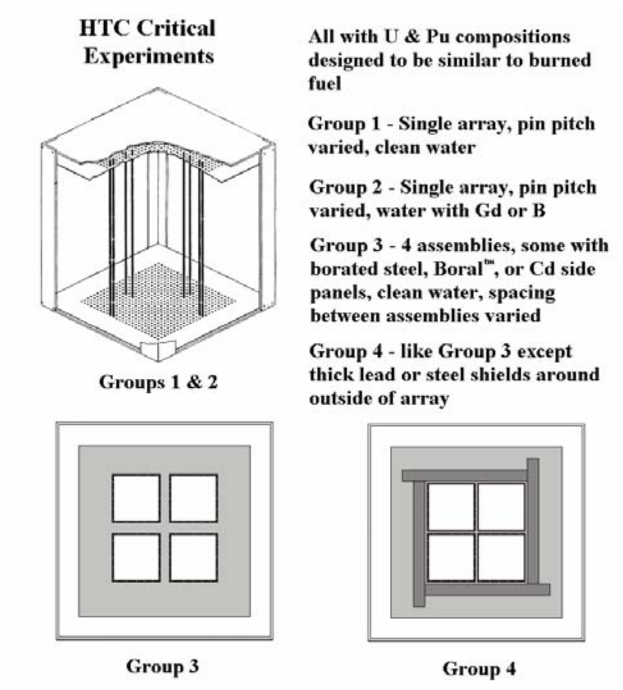

Figure 2. French HTC critical experiments.

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.[11]

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 that ORNL is seeking to obtain from Cogema. ORNL has received preliminary reports that describe 147 critical configurations (referred to as the "PF" experiments), 74 of which contain fission products. In the ensuing months ORNL will work to perform S/U analyses for these French fission product experiments using TSUNAMI-3D and TSUNAMI-IP. From a review of the preliminary experiment descriptions, it is anticipated that the c_k values will be lower than 0.8 for many; however, ORNL is working to use TSUNAMI-IP to estimate the uncertainty allowance that can be added based on the use of the sensitivity profile comparison and a propagation of uncertainty information on the nuclear data.

III.C. Assessment of Commercial Reactor Critical Configurations

Work is in progress to perform S/U analyses for more than 60 CRC state points. The initial focus has been on the reactor core configurations and material compositions for 33 Crystal River Unit 3 state points.[12,13] 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 three of the Crystal River CRC state points show c_k >0.85 for CRC cases with effective full power days ranging from 0 to 515. In addition, a comparison of the sensitivity files show reasonable similarity for many of the key fission products. Work is continuing to analyze all of the available CRC state points and assess their utilization in burnup credit criticality evaluations.

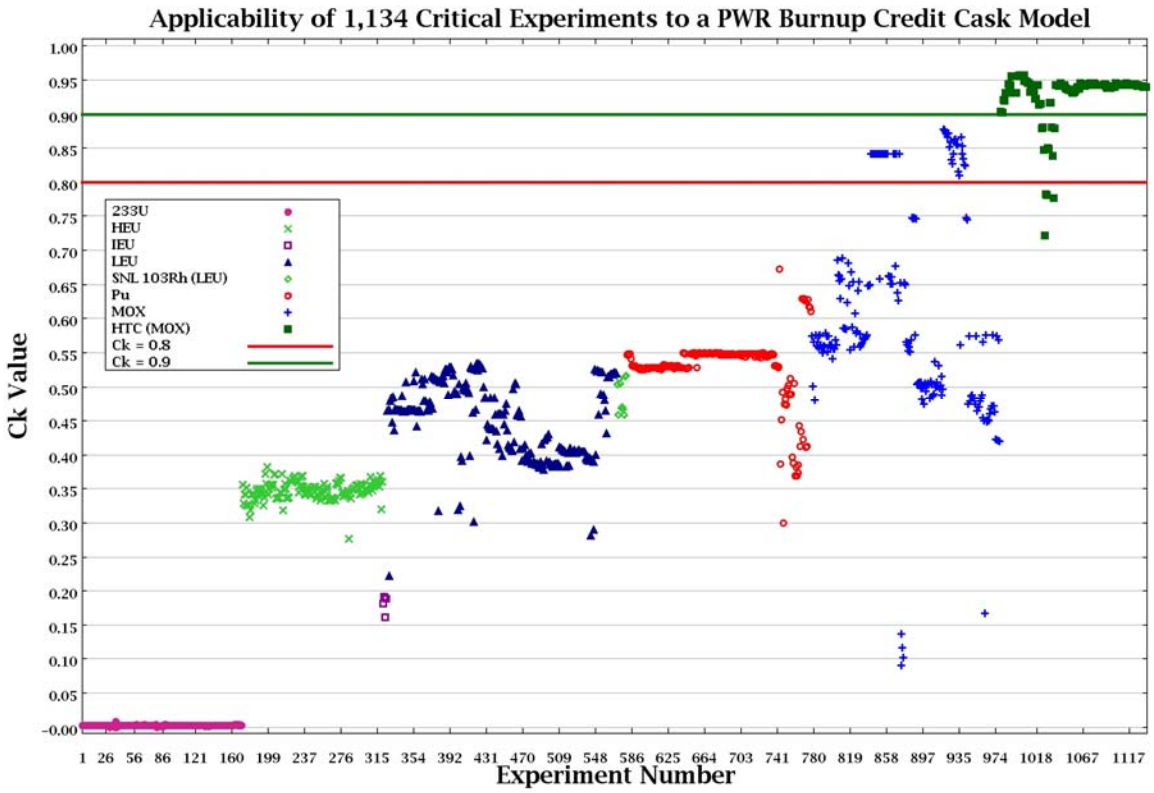

Figure 3. Critical experiment applicability to burnup credit.

III.D. Proposed New Critical Experiments

This joint project is seeking to pursue all existing options to help bring closure to the current technical issues related to burnup credit. To this end, the project is pursuing planning activities to perform additional experiments with the principal fission products. The experiments are to be performed at SNL in a manner similar to the critical experiment 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 in 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 Cogema (e.g., data that might be needed to address burnup credit for BWR SNF). Initial planning activities were initiated in 2005, with critical experiments expected to begin sometime in 2007.

IV. DATA BASE 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 or decay time, the top six 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 are the focus of this project's efforts to obtain and assess potential sources of 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) [Ref. 14]. Of the 56 PWR spent fuel samples that had been evaluated by ORNL prior to 2005 [Ref. 6], only 19 included 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.[15] Table II provides a summary of the total number of measurements assessed and accepted 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)[16] and the Japanese Takahama Unit 3 PWR fuel measurements performed by the Japan Atomic Energy Research Institute.[17]

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 data base 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. Statistically, the uncertainty is best estimated if at least 15 to 20 measured samples are available; the project goal is thus to have this minimum number of measurements available for the validation of the principal fission product nuclides.

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'Energie Atomique (CEA) of France has established experimental programs to provide data for the validation of French computer codes. The programs include extensive spent fuel assay measurements in support of fuel inventory and fuel cycle studies, including burnup credit.[10] The data from these programs are proprietary but 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 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 assay measurements include three SNF samples with 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 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-accuracy radiochemical analysis methods employed, (2) the wide range of enrichments and burnups (covering most commercial U.S. fuels), (3) the use of standard commercial fuel assemblies (nonreconstituted), 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 about 20.

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

(Highest 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. Through support from NRC and DOE, ORNL is participating in both of these programs, which will provide fission product assay data measured by several independent laboratories using state-of-the-art methods. The REBUS program will provide fission product assay data for one PWR SNF sample, while the MALIBU program will provide fission product assay data for two PWR SNF samples. However, the number of assay samples that are being evaluated is small. The data will be proprietary for 3 years after the final report is issued. These data will be evaluated by ORNL in 2006 and included in a publicly distributed data base at the end of the 3-year proprietary period.

IV.C. Sources of Additional Assay Data—Nonproprietary

In 2005, ORNL contracted with PNNL to investigate and assess whether there are existing, U.S.-origin SNF samples that can be retrieved and made available for expanding the data base of radiochemical assay data for validation of fission product burnup credit. A large percentage of the existing usable fission product assay data was generated by the Material Characterization Center (MCC) at PNNL as part of the ATM program in the late 1980s and early 1990s. ORNL has received a draft report from PNNL identifying available samples. ORNL plans to evaluate the need for performing measurements on some or all of these samples.

A major activity has been work to reassess reported measurements of Three Mile Island Unit 1 (TMI-1) SNF that were performed circa 1999 to support the YMP.[18] An earlier assessment of the TMI-1 data by ORNL deemed the TMI-1 data were not suitable for use in obtaining the bias and uncertainties for prediction of fission product nuclides. The basic reason for this conclusion was that analyses performed by both ORNL and staff at the YMP[19] showed the C/E results to be highly discrepant compared with the results from the other 56 samples analyzed by ORNL and those reported by the CEA and Belgonucleaire programs. 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 reactor physics codes (both SCALE and HELIOS) show discrepancies of 10–20%. This compares with typical calculated-to-measured differences of ±5% for ^{239}Pu. 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.[20] The reactor conditions experienced by these fuel samples are not well known. Several suspected local conditions[20] that could significantly impact the predictions could explain the larger C/E discrepancies.

Nevertheless, the difficulty in obtaining the quantity and quality of measured assay data for fission product nuclides has led ORNL to revisit the potential usefulness of the TMI-1 data. There are 19 TMI-1 measured samples having a desirable range of initial enrichments (4.0–4.65 wt %) and burnup values (23–55 GWd/MTU). Thus, the TMI-1 samples provide the number of additional measurements recommended for adequate statistical estimation of the uncertainties. The supposition is that a number of samples of "poor" quality (high bias and uncertainty caused by unknown factors) might be similar to a small number of samples deemed to be of high quality (accurate radiochemical measurements with well-known reactor conditions). Thus, ORNL has recently investigated the distribution of the TMI-1 C/E values and carefully studied the available information on the TMI-1 reactor conditions for this fuel.

The initial recommendation from this reinvestigation, pending further work in 2006, is that the TMI-1 samples should not be considered sufficiently qualified for code benchmark purposes (demonstrating that the code and its input data are accurately predicting reality). However, the samples may be useful in supporting a safety basis, provided that the uncertainties are adequately addressed and that use of the data can be demonstrated to yield conservative results. To demonstrate that use of the TMI-1 data provides conservative results requires, at a minimum, a few high-quality measurements from other sources. For fission product nuclides having no previous measurements (e.g., ^{95}Mo, ^{101}Ru), it will be difficult to establish that the TMI-1 results are representative or conservative without having independent data. Also, with any use of the TMI-1 data, it must be recognized that the uncertainties derived from the data may not be representative of modern high-burnup fuel. Ultimately, it should be demonstrated that use of the data does not reduce the margin because of the addition of data that may exhibit abnormal biases. Some additional work in this area is expected prior to final recommendations. The outcome of this work may also influence the effort expended under this project to obtain proprietary data or additional domestic assay data.

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. Under this project, ORNL is working 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). As needed and justified, new measurements will be performed under a cooperative DOE-Euratom agreement. Work has already been initiated on new measurements and evaluation for ^{103}Rh. Production cross-section libraries will be prepared that are consistent with the quality and rigor now provided in the actinide data.

VI. OTHER ACTIVITIES

VI.A. Data for Improved Safety Analyses

ORNL used a summer intern to gather and organize operational parameter data from PWR and BWR CRC information to support establishment of more-realistic bounding assumptions for use in the safety analyses. Soluble boron concentrations, maximum fuel temperature, and minimum moderator densities were the initial parameters investigated. Using the range of data values obtained and investigating the mean standard deviations, ORNL is working to provide a technical basis for recommending bounding assumption values that can be used in the safety analysis. A reduction in the conservative values recommended in earlier reports is anticipated; the reduction should allow a larger fraction of spent PWR fuel to be considered as acceptable for transport in fully loaded high-capacity casks. This activity is a continuing effort.

VI.B. BWR 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 de-rated (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. A simplistic cost savings analysis, based on reduction in the number of shipments, for BWR burnup credit was performed. This cost savings analysis and the work to date on BWR burnup credit will be documented in 2006. Simple and reliable approaches for using burnup credit to assure full cask loadings of all inventory up to 5 wt % will also be explored.

VII. SUMMARY

This report has summarized the activities performed by this project to date. A simple, but straightforward approach for quantifying the benefits of PWR fission product burnup credit was developed and can be extended to various transport scenarios as needed. The assessment 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 set in final form and the PF or fission product critical experiment set in draft form) 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 all known sources of assay data and initiated a new effort to reassess and provide guidelines on utilizing the TMI-1 measured data that provide large and atypical C/E values relative to all other known sources of data.

ORNL also has continued to seek a diverse path in assuring that all technical approaches are studied and understood to (1) provide flexibility in future safety analyses and (2) ensure that a solid technical basis consistent with cost and benefit is established. Thus, the CRC 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 data base 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). By the end of 2006, ORNL is hoping to provide NRC with draft recommendations on implementing fission product credit using the data that have been obtained and to demonstrate where future work (e.g., planned experimental data or an improved data base on reactor operating history) might improve implementation of full burnup credit.

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