Cooperative Project on Burnup Credit—Data and Analysis for Spent Nuclear Fuel Transport and Storage in Burnup Credit Casks

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This report describes research sponsored by the Electric Power Research Institute (EPRI), the U.S. Department of Energy Office of National Transportation, and the U.S. Nuclear Regulatory Commission.

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ABSTRACT

A key part of EPRI's research is directed towards assuring efficient and effective management of spent nuclear fuel (SNF) from commercial nuclear power plants. A summary of the 2005 burnup-credit-related activities jointly sponsored by the U.S. Department of Energy Office of National Transportation, the U.S. Nuclear Regulatory Commission, and the Electric Power Research Institute (EPRI) is presented.

A simple, but straightforward, approach for quantifying the benefits of PWR fission product burnup credit was developed. The assessment indicated a savings in transport costs alone in the range of $150 – 400M.

A contract with Cogema to gain access to the results of the extensive experimental program conducted in France in support of burnup credit was successfully negotiated. The highest priority data have been obtained (the HTC, or high burnup, 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 spent nuclear fuel (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. Oak Ridge National Laboratory (ORNL) has investigated all known sources of assay data and initiated a new effort to re-assess and provide guidelines on utilizing the TMI-1 data, which provide large and atypical 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) assure 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 to increase the database via domestic (e.g., new critical experiments at SNL and assay data measurements at PNNL) or international (e.g., participation in international research programs) participations are ongoing.

By the end of 2006, ORNL is expected to be in a position to provide draft recommendations on implementing fission product credit using the data that have been obtained and demonstrate where future work (e.g., planned experimental data or an improved reactor operating history database) might improve implementation of full burnup credit.

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1 INTRODUCTION

A key part of EPRI's strategic research is directed towards assuring efficient and effective management of spent nuclear fuel (SNF) from commercial nuclear power plants. Spent fuel cask designs 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-y cooling time for transport of SNF, it became quickly apparent in the late 1980s that SNF cask capacity would often be limited by the conservative, yet simple fuel assumption of un-irradiated 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 rail casks by at least 30%.

The use of high-capacity casks leads to reduced risk and reduced cost relative to storage and transport operations. 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 only recommends credit for the reactivity change due to major actinides (reduction in actinides that fission and increase in actinides that absorb neutrons). The current regulatory position [1] for transport and storage is provided in the Nuclear Regulatory Commission's (NRC's) Interim Staff Guidance 8, Revision 2 (ISG-8r2). This guidance only allows for actinide-only burnup credit and will enable only about 30% of the domestic SNF inventory from PWRs to be loaded in high-capacity (~32 PWR assemblies) casks. Additional burnup credit provided by fission products (nuclides produced during burnup with neutron-absorbing properties) is necessary to enable high-capacity casks to handle the majority (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 EPRI, the NRC, and the Department of Energy (DOE) Office of National Transportation (ONT) to obtain the funding needed 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 use of burnup credit in boiling water reactor (BWR) SNF casks will be investigated with the goal of recommending the technical approach and associated data needs for BWR fuel with enrichments up to 5 wt% to be transported in high-capacity casks.

Although funding from EPRI has been directed at procuring experimental data, this report provides the progress on all aspects of this cooperatively-funded, multi-year project through October 2005.

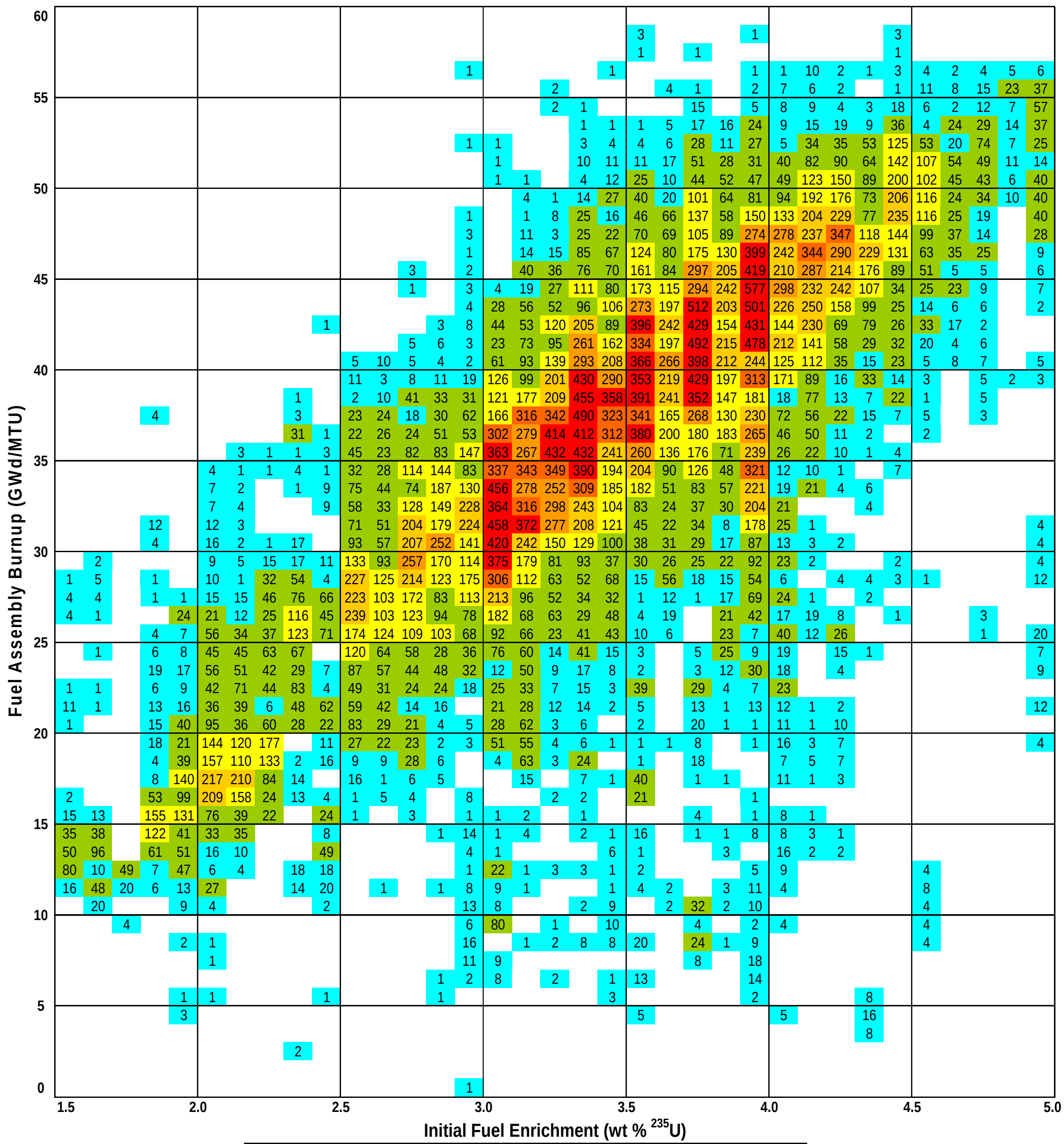
2 ASSESSMENT OF BENEFITS FOR FULL BURNUP CREDIT

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 (see Figure 1). Six of the PWR fuel assembly types - WE 17 × 17, WE 15 × 15, WE 14 × 14, B&W 15 × 15, CE 16 × 16, and CE 14 × 14 - comprise about 94% of the 70,290 PWR SNF assemblies in the data base. These six types of PWR assemblies were investigated to assess the benefits that would be provided by full burnup credit.

A review of the RW-859 (2002) data reveals that the average burnup of discharged PWR fuel assemblies has risen from around 20 GWd/MTU in 1975 to 45.7 GWd/MTU in 2002. This increase in assembly average burnup represents a significant increase in the amount of criticality safety margin potentially available through burnup credit. Through 2002, 18.1% of the 70,290 discharged PWR fuel assemblies had burnups greater than 45 GWd/MTU. The average initial ^{235}U enrichment of discharged PWR assemblies has risen from about 2.7 wt% in 1975 to 4.2 wt% in 2002. This trend of increasing initial enrichment has made the fresh fuel assumption typically used in criticality safety analyses a more restrictive approach for cask design.

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.



Number of Assy in Group			Total Assy	% of All	Cum. Assy
0	to	20	3,385	4.8	70,128
21	to	100	14,937	21.3	66,743
101	to	200	15,082	21.5	51,806
201	to	250	10,891	15.5	36,724
251	to	300	6,354	9.1	25,833
301	to	350	5,555	7.9	19,479
351	to	577	13,924	19.9	13,924

Figure 1. PWR Spent Fuel Inventory from RW-859 (2002) Nuclear Data Files

Loading curves, consistent with the regulatory guidance of Ref. 1, are provided in Figures 2 and 3 for two of the six assembly types. The acceptability of the SNF assemblies for each fuel type is summarized in Table 1. 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 can enable loading a large percentage of the CE 14 × 14 and WE 14 × 14 assemblies in a high-capacity cask, its effectiveness under the current regulatory guidance is minimal for the other assembly designs considered.

Table 1. 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%)

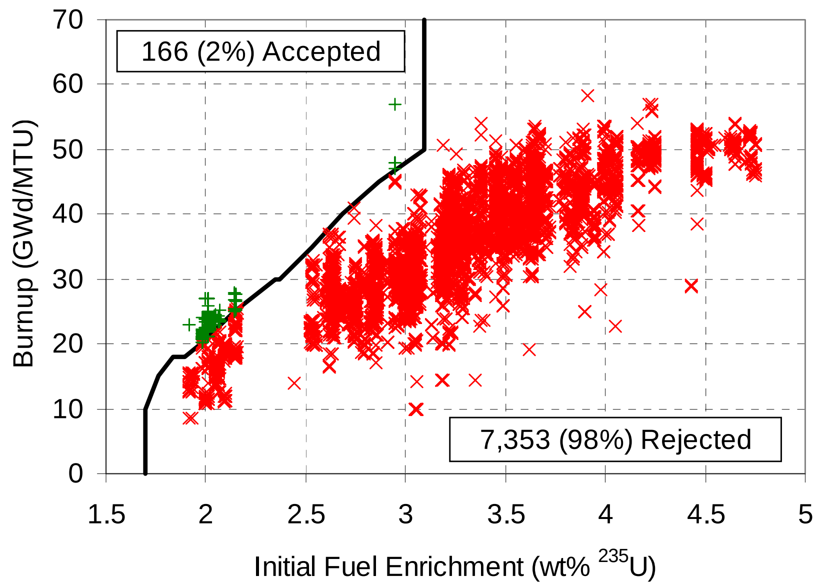


Figure 2. B&W 15 × 15 Inventory Shown with ISG-8r2 Burnup Credit Limit Curve

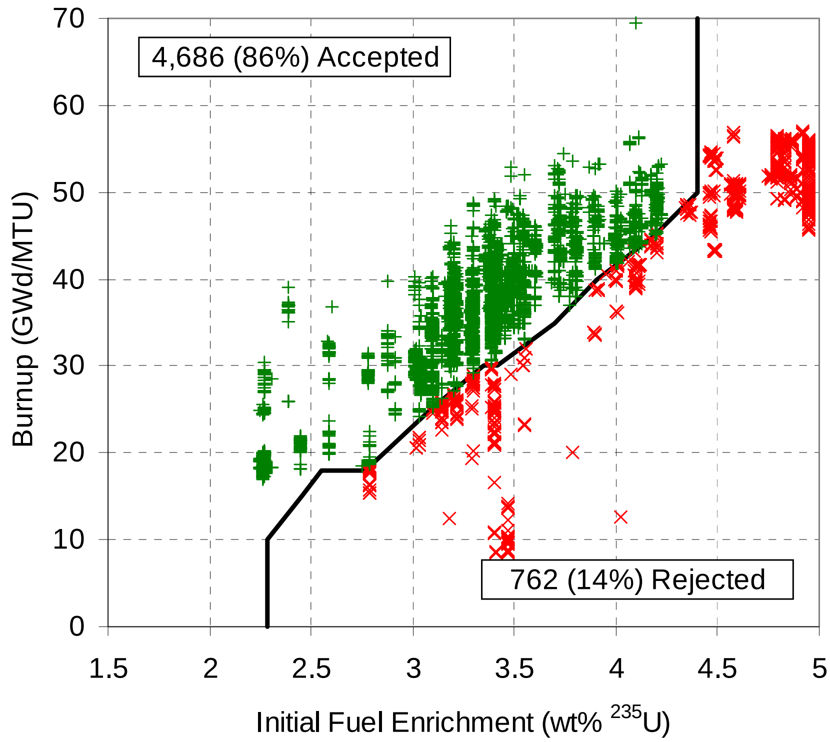


Figure 3. WE 14 × 14 Inventory Shown with ISG-8r2 Burnup Credit Limit Curve

To evaluate the effect of selected calculational assumptions, Figure 4 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 5 of the principal 6 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 (^{103}Rh is excluded due to insufficient measured assay data);

(2) Inclusion of minor actinides and 5 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; and

(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 is available to estimate bias and uncertainty. Thus, with the exception of the final case, no credit was taken for their presence in the SNF.

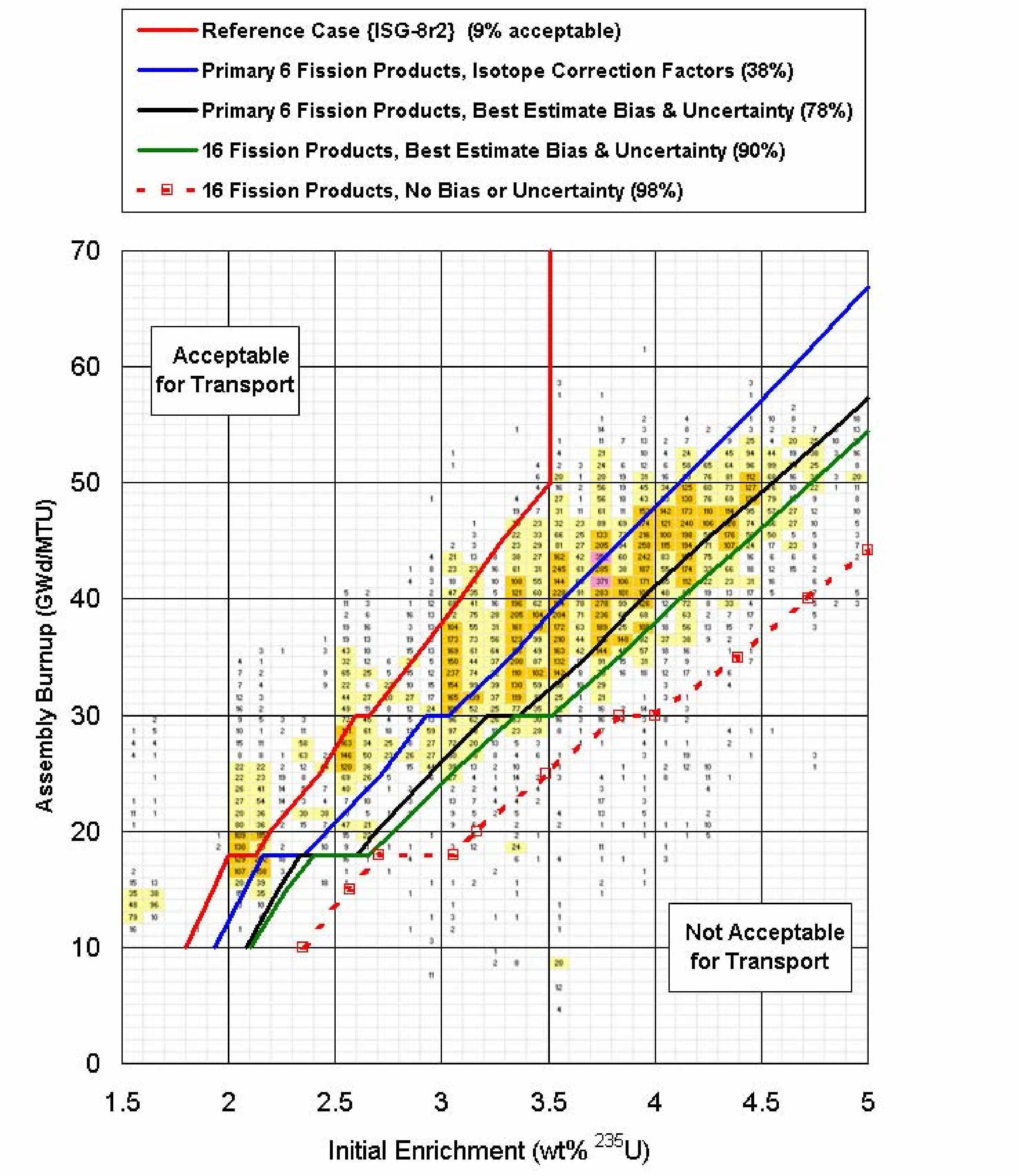


Figure 4. Comparison of Calculational Assumptions for WE 17 × 17 Fuel Assemblies. (Percentages of inventory acceptable for the GBC-32 cask are shown in parentheses)

All of the curves in Figure 4 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 4 where inclusion of fission products and/or the use of more-realistic approaches to isotopic validation offers 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 primary 5 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 4 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. For all the cases with fission products included, no explicit consideration of reactivity bias and uncertainty from comparison with critical experiments is included. However, the loading curves are all based on an upper subcritical limit of 0.94 (as opposed to 0.95), which inherently allows 1% Δk for criticality calculational bias and uncertainty.

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 (through either increased credit for fuel burnup or cask design/utilization modifications) is necessary to accommodate the majority of PWR SNF assemblies in high-capacity casks. The loading curves presented in this report are such that a notable portion of the SNF inventory would be unacceptable for loading because the burnup value is too low for the initial enrichment. 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.

Cost Benefits for PWR SNF

An initial economic analysis of burnup credit for transportation was prepared for DOE/RW in 1988 and used a life cycle cost model to estimate a potential savings up to $900M. [7] Since that time, a portion of this predicted savings has become obtainable via the actinide-only credit allowed by ISG-8r2. Under this project, a relatively simple, but more current cost analysis of the potential benefits of burnup credit was initially completed in 2003. The analysis used the current 70,000 Metric Ton Heavy Metal (MTHM) capacity limit for the Yucca Mountain repository, the percentage of total MTHM from PWRs as of 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 assemblies that cannot be accommodated in a 32-assembly cask are transported in a 24-assembly cask, 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 experienced opinions leaned toward the $500k/shipment value, a conservative estimate of $250K was adopted. Using this per-shipment estimate provides a resulting cost savings (assuming shipments reduced by 625 = 940 - 315) 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 (i.e., 100-125 ton) rail-type casks. In 2005, the cost estimate was updated to remove the simplifying assumption and investigate the impact of using a cask fleet of varying sizes. Discharge data as a function of site capabilities was first obtained (see Table 2). For the various cask sizes that could be used, estimates for (1) cost per cask shipment; (2) cask design capacities with and without burnup credit; and (3) percentages of assemblies acceptable for loading with and without burnup credit were developed. These estimates are listed in Table 3. Using the discharge data from Table 2 and the analysis assumptions listed in Table 3, the cost savings associated with burnup credit for transportation is estimated in Table 4 to be ~$638M; ~$235M of which is attributable to credit for fission products. These estimates are consistent with the previous analysis and demonstrate the significant potential cost savings associated with establishing burnup credit that includes credit for the fission product compositions. The results are based solely on cost savings associated with the reduction in the number of shipments for PWR SNF; cost savings associated with reduced personnel dose, public exposure, and accident risks are not included.

Limited sensitivity analyses were performed to evaluate the sensitivity of the cost savings estimates to variations in the input assumptions listed in Tables 2 and 3. In general, it was found that increased use of smaller casks will increase the cost savings. This trend is shown in the last column of Table 4, which lists savings due to fission product burnup credit on a per-assembly basis, and is due to the increased shipment cost on a per-assembly basis associated with the use of smaller casks. Assuming all 113,109 assemblies are transported in anyone of the various cask sizes yields a range of $177-424M in estimated cost savings attributable to fission product burnup credit, with the lowest number corresponding to the use of all large rail-type casks and the highest number corresponding to the use of all truck casks. Note that the assumptions listed in Table 3 account for the fact that the increase in the percentage of acceptable assemblies due to fission product burnup credit is much less for smaller casks.

Although this most recent analysis does not specifically address decay heat constraints (e.g., if utilities opt to transport hottest fuel first) that could require a reduction in capacity for the large rail-type casks, it does show that the use of smaller casks (e.g., to transport SNF with high decay heat) results in greater cost savings when burnup credit is applied. Also, there is a considerable portion of the discharged SNF inventory that will not present challenges in terms of decay heat and the ability to use full burnup credit will provide a significant degree of flexibility to the vendors and utilities seeking to optimize their cask loadings.

In conclusion, the assessment performed under this project has shown the estimated cost savings associated with extended burnup credit is greater than $150M and is most likely in the $200-300M range. Evaluation of the variations in the relevant input assumptions used to develop these estimates provides confidence that the actual cost savings may be much higher, but are not likely to be lower.

Table 2. Number of Projected Discharged SNF Assemblies as a Function of Site Capability

Cask size code	Range of site capabilities (tons)	Number of assemblies[1]
LWT	LWT ≤ 25	3,234
OWT	25 < OWT ≤ 35	4,734
RC1	40 < RC1 ≤ 75	8,443
RC2	75 < RC2 ≤ 100	52,333
RC3	100 < RC3 ≤ 125	36,426
RC4	125 < RC4	7,939
	Total	113,109

Table 3. Analysis Assumptions for the Various Cask Sizes

Cask size (tons)	Cost/ Shipment ($k)[2]	Design capacity (no. of assemblies)[3]		Fraction of assemblies acceptable for loading[4]		
		w/o BUC	w/ BUC	w/o BUC	w/ AO[5] BUC	w/ AFP[6] BUC
LWT ≤ 25	150	2	4	1	0.9	1
25 < OWT ≤ 35	200	4	6	1	0.8	1
40 < RC1 ≤ 75	200	7	10	1	0.7	1
75 < RC2 ≤ 100	200	12	18	1	0.5	0.9
100 < RC3 ≤ 125	250	24	32	1	0.3	0.9
125 < RC4	250	24	32	1	0.3	0.9

[1] Data corresponds to the number of assemblies discharged through 12/31/1998 plus those projected to be discharged through 12/31/2015 (source: RW-859).
[2] Values are intended to include freight, operational, and security costs and are based on a review of industry experts/experience and information generated during the process of evaluating the use of dedicated trains. The latter source suggested a cost of ~$200k per cask shipment for freight and security only; no estimate of operational cost was available.
[3] Values developed based on a review of published and unpublished information, as well as consultation with industry experts.
[4] Values based on specific analyses, published results, and analytical experience.
[5] "AO BUC" refers to burnup credit that only accounts for the principal actinide compositions, consistent with current regulatory guidance (ISG-8r2).
[6] "AFP BUC" refers to burnup credit that includes the principal actinide and fission product compositions. This is also referred to as "full" burnup credit, which is not permitted under current regulatory guidance (ISG-8r2).

Table 4. Summary of Cost Savings

Cask size code	Number of assemblies	Number of shipments			Cost savings ($1k)		Savings due to FP BUC ($1k)	
		w/o BUC	w/AO BUC	w/AFP BUC	w/AO BUC	w/AFP BUC	total	per assembly
LWT	3,234	1,617	889	809	109,200	121,200	12,000	3.71
OWT	4,734	1,184	868	789	63,200	79,000	15,800	3.34
RC1	8,443	1,206	953	844	50,600	72,400	21,800	2.58
RC2	52,333	4,361	3634	3053	145,400	261,600	116,200	2.22
RC3	36,426	1,518	1404	1176	28,500	85,500	57,000	1.56
RC4	7,939	331	306	256	6,250	18,750	12,500	1.57
Totals	**113,109**	**10,217**	**8,054**	**6,927**	**403,150**	**638,450**	**235,300**	

3 DATA BASE OF CRITICAL EXPERIMENTS FOR FULL BURNUP CREDIT

Background and Approach

To achieve the potential benefits discussed and demonstrated in Section 2, this project is seeking to obtain the data needed for preparation and review of a criticality safety evaluation with full burnup credit. The rationale for restricting the current regulatory guidance for burnup credit to actinide-only is largely based 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. Even for actinide-only burnup credit, there is a need to access a wide spectrum of existing critical experiments to properly validate the analysis methods for estimating the reactivity and understand the uncertainties. Thus, a patchwork approach is needed to assure that the fuel compositions, fuel geometry, and cask-like configuration are all properly considered in the validation. There is also a need to demonstrate that the critical experiments, selected from an existing set of experiments not intended to address burnup credit, are indeed applicable to the cask design and fuel condition. In summary, applicants and regulatory reviewers are hampered by both a scarcity of data and a lack of clear technical bases (e.g., criteria) for demonstrating applicability of the data.

The difficulty of the patchwork approach and the scarcity of applicable experiment data are shortcomings that were a consideration in the NRC decision to limit their regulatory guidance to actinide-only. Although the quantity of fission product credit available for a particular fuel and cask design can not be validated, it is a scientific fact that fission products are neutron absorbers and the presence of the fission products reduces the reactivity. By ignoring the presence of the fission products, actinide-only burnup credit provides additional reactivity margin that NRC has judged to adequately compensate for any additional uncertainty that may not be identified from the sparse set of relevant experiment data applicable to actinide-only burnup credit.

Under this project, ORNL is working to obtain, and make available to industry, a well-qualified experimental data base that can assure 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*) [8],
2) Proprietary critical experiment data,
3) Commercial reactor criticals (CRCs), i.e., critical state-points from operating reactors,

4) Proposed new critical experiments.

The applicability and value of this data base of critical experiments is being assessed using sensitivity and uncertainty (S/U) analysis tools developed at ORNL and incorporated within Version 5 of the SCALE code system. [5, 9] The TSUNAMI-3D sequence within SCALE uses first-order linear perturbation theory [10] 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 parameter, referred to as c_k, which is a single-valued parameter used to assess similarity of uncertainty-weighted sensitivity profiles between a modeled system and a criticality experiment for all nuclide-reactions. A c_k parameter 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 means the systems are completely dissimilar. The current guidance [9] 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 (see Fig. 5) 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.

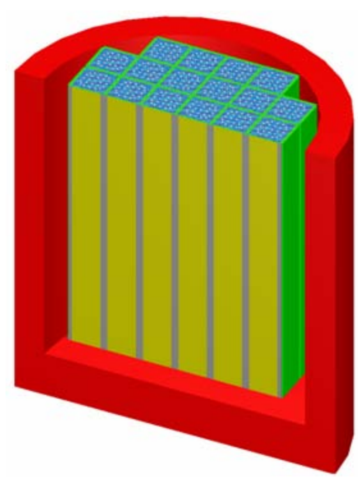


Figure 5. GBC-32 Cask Model

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 [11] to develop a technical basis for burnup credit. Subsequent to assessment and evaluation, data obtained by ORNL under the contract will be made available to industry for use in cask design and licensing activities.

In late July 2005, ORNL received the first set of critical experiment data documented using the format of the IHECSBE. These experiments were performed with rods having U and Pu isotopic compositions similar to U(4.5%)O_2 fuel with a burnup of 37,500 MWd/MTU. The experiment series is referred to as the HTC experiments and there are 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 is dissolved in the water at 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.

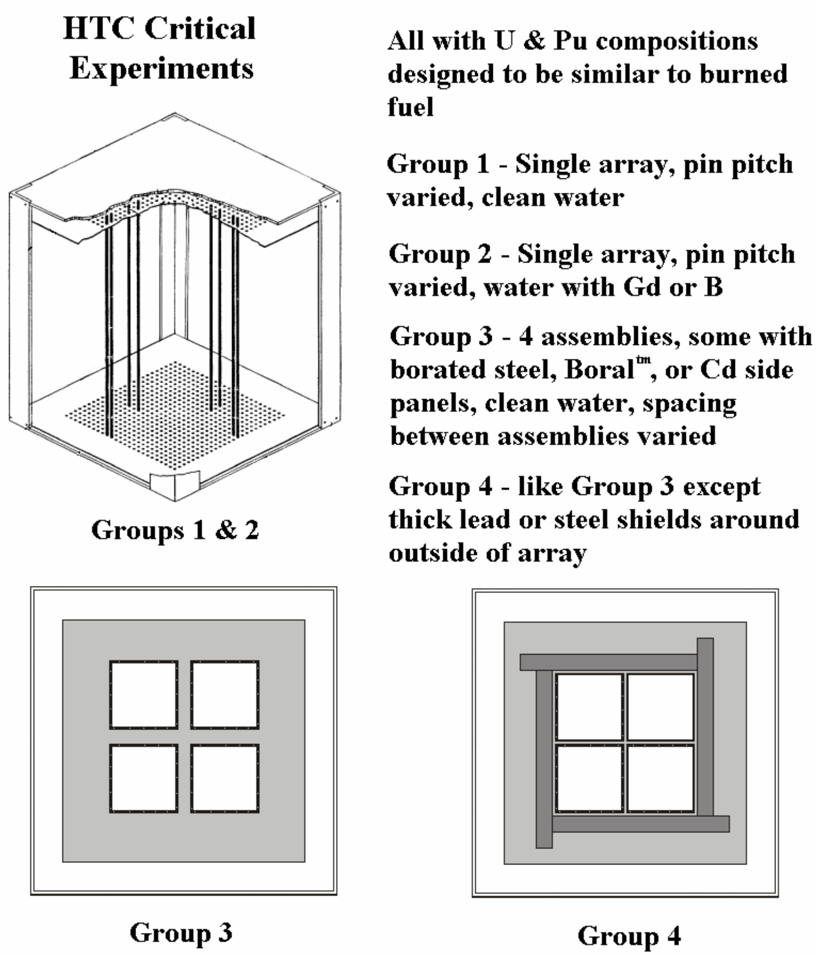


Figure 6. French HTC Critical Experiments

These 156 HTC critical experiments, together with nearly 1,000 critical configurations from the IHECSBE, have been analyzed with the TSUNAMI-IP sequence and the sensitivity data obtained has been compared with sensitivity data for the reference cask model loaded with assemblies burned to 40 GWd/MTU (actinides and fission products are included in the reference model). Figure 7 shows the distribution of the c_k values for the 1,134 critical configurations when compared to the reference burnup credit cask model. As shown in the figure, the 170 ^{233}U experiments, the 150 high enrichment uranium, the 4 intermediate enrichment uranium, 197 plutonium-only configurations, and the 256 low enrichment uranium, all have c_k values < 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 substantially weaker validation basis that exists without the HTC experiments.

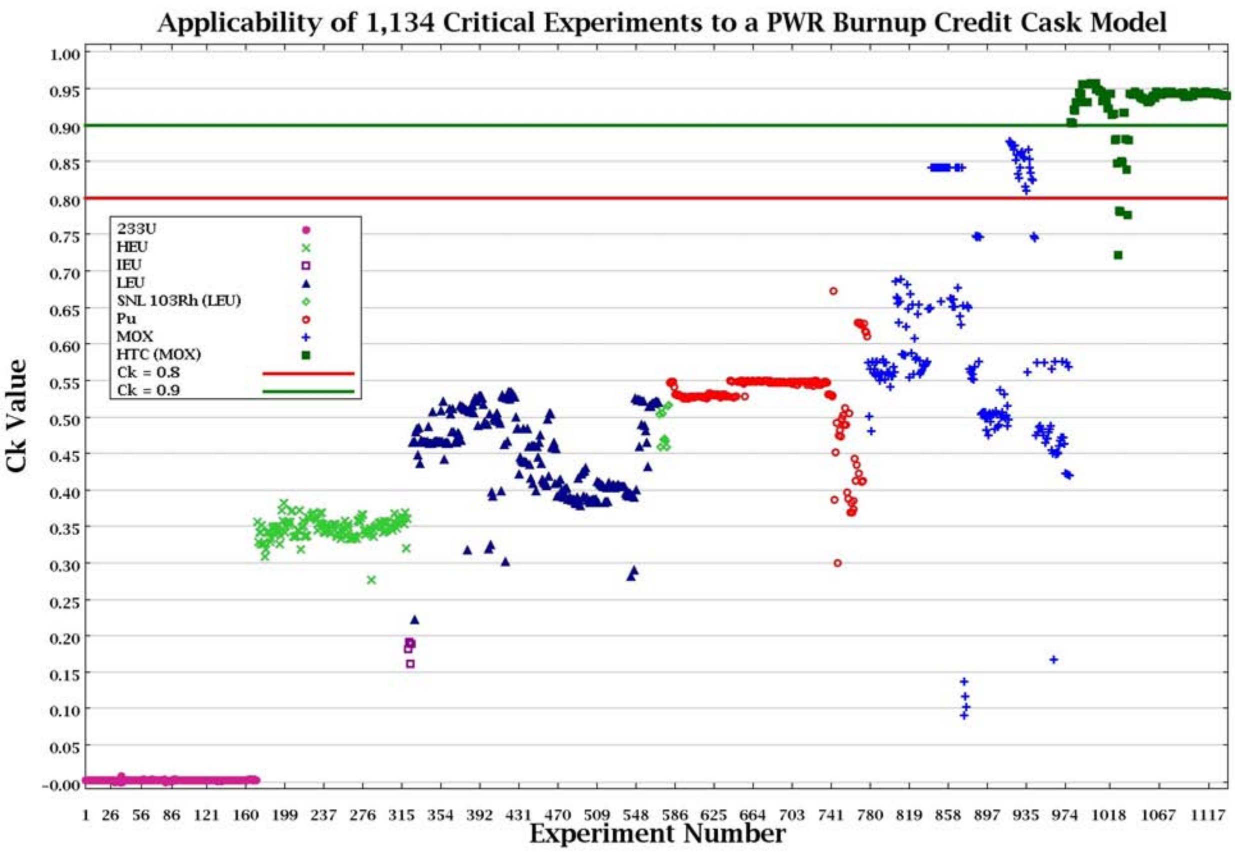


Figure 7. Critical Experiment Applicability to Burnup Credit

Work has been initiated to assess critical experiments for validating the fission product component of SNF in a cask environment. In 2005, work was performed to assess two sets of critical experiments involving fission products. 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. Under this current project, the final documentation and review of these experiments were completed and published as part of the 2005 release of the IHECSBE data base.

Sensitivity/uncertainty 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. Figure 8 shows how the ^{103}Rh from the SNL experiments compares with the ^{103}Rh in the GBC-32 cask. The coverage is reasonably good 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 (see Section 5) to ensure maximum applicability. [12] Although the c_k values for these experiments are lower than 0.8, the goal is 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.

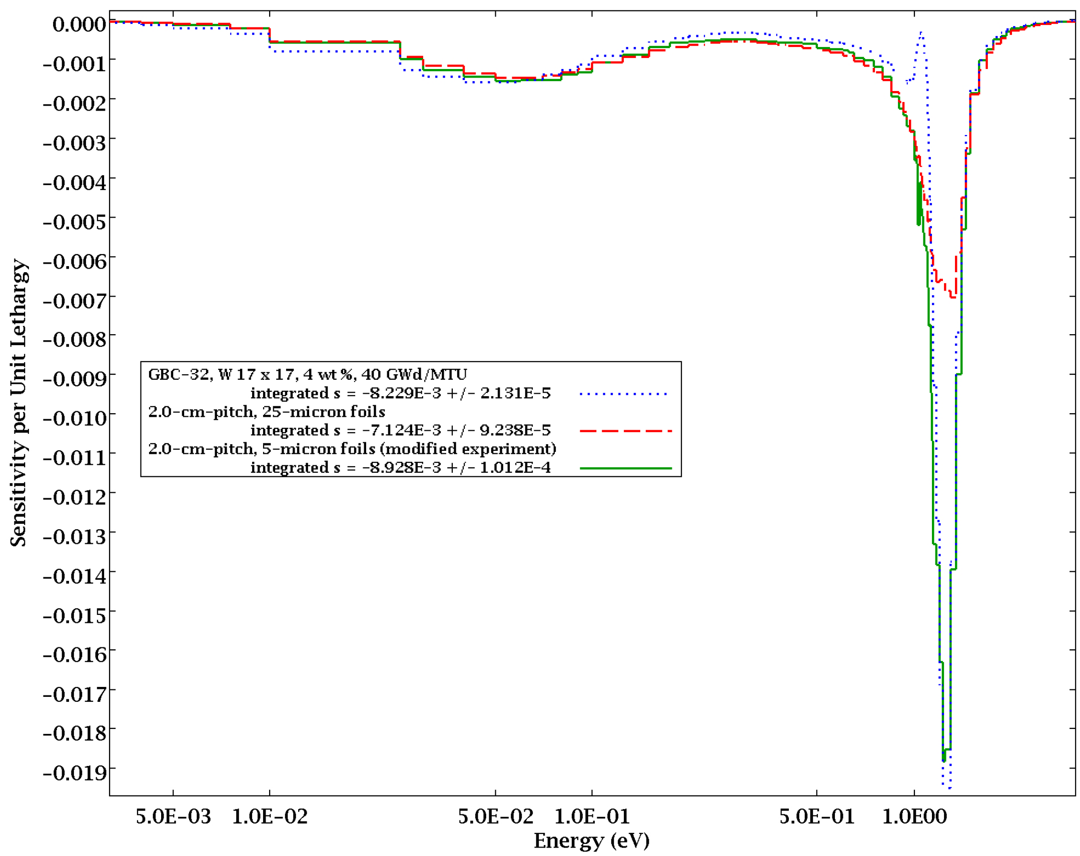


Figure 8. Comparison of ^{103}Rh Sensitivity Profiles from the GBC-32 Cask and the SNL ^{103}Rh Critical Experiments

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 via the contract noted above. ORNL has received preliminary reports that describe 147 critical configurations (referred to as the "PF" experiments), 74 of which contain fission products. The HTC critical experiment MOX rods were used in 29 of the critical configurations and 14 of these contained fission products. The fission products were present in solution either individually or as mixtures. The first group of experiments has a central tank filled with water, borated water, or fission product solution. The central tank is surrounded by U(4.7)O2 fuel rods in water. The second group of experiments has a central tank containing an 11x11 array of either U(4.7)O2 or HTC MOX rods in uranyl nitrate solutions with dissolved fission products The central tank is surrounded by U(4.7)O2 fuel rods in water. The third group of experiments has a large tank containing an array of either U(4.7)O2 or HTC MOX rods in depleted uranyl nitrate solutions. Four of the Group 3 experiments with HTC MOX rods also contain fission products. In Group 3, the tank is surrounded by water. Preliminary sensitivity analyses of these French fission product experiments using TSUNAMI-3D and TSUNAMI-IP indicate that only four of the 147 critical configurations are similar enough to the GBC-32 cask model to yield c_k values greater than 0.8. These four configurations are nearly identical to each other and yield c_k values of about 0.97. Using TSUNAMI-IP, the goal for early 2006 is to quantify an uncertainty allowance for the fission products by using the sensitivity profile information and the limited number of applicable critical configurations that have high c_k values.

Assessment of Commercial Reactor Critical Configurations

Work currently in progress includes modeling and S/U analyses for more than 60 Commercial Reactor Critical (CRC) state-points. The initial focus has been on the reactor core configurations and material compositions for 33 Crystal River Unit 3 state-points that are documented in great detail in the Yucca Mountain Project (YMP) reports [13, 14]. In addition, the YMP reports document SCALE/SAS2H [5] and MCNP [15] calculations for the 33 state-points. The technical information provided includes 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. The CRC state points require very large, complex computational models and application of the S/U tools further extends the complexity, and therefore time and effort needed to prepare the models for analysis.

Due to the large amount of information available for the Crystal River Unit 3 critical state-points and to ensure accurate modeling, a computer program has been developed to automatically prepare SCALE/CSAS25 [5] and TSUNAMI-3D input files. The first complete version of the computer program for automatic generation of SCALE/CSAS25 input files has been completed and is undergoing initial testing. The current version of the program writes the material composition data block, lattice cell data block, and SCALE/KENO V.a geometrical units describing all reactor assemblies including assemblies that contain BPRA, RCCA, and APSRA. Initial testing/verification of the input files generated by the program indicated the need for some revisions to the conversion code to eliminate redundant materials from the models and thereby produce more efficient computational models (needed for TSUNAMI analyses). Figure 9 shows an overhead view of the Crystal River Unit 3 model as generated by the SCALE graphical display package. Initial criticality analyses have produced expected k_{eff} values from the SCALE/CSAS25 sequence and work to obtain the S/U results are expected by the end of 2005. The S/U analyses will provide c_k results comparing the CRC models and the GBC-32 cask loaded with SNF. The goal for early 2006 is to identify the CRC characteristics that are most applicable to validation of SNF in transport casks.

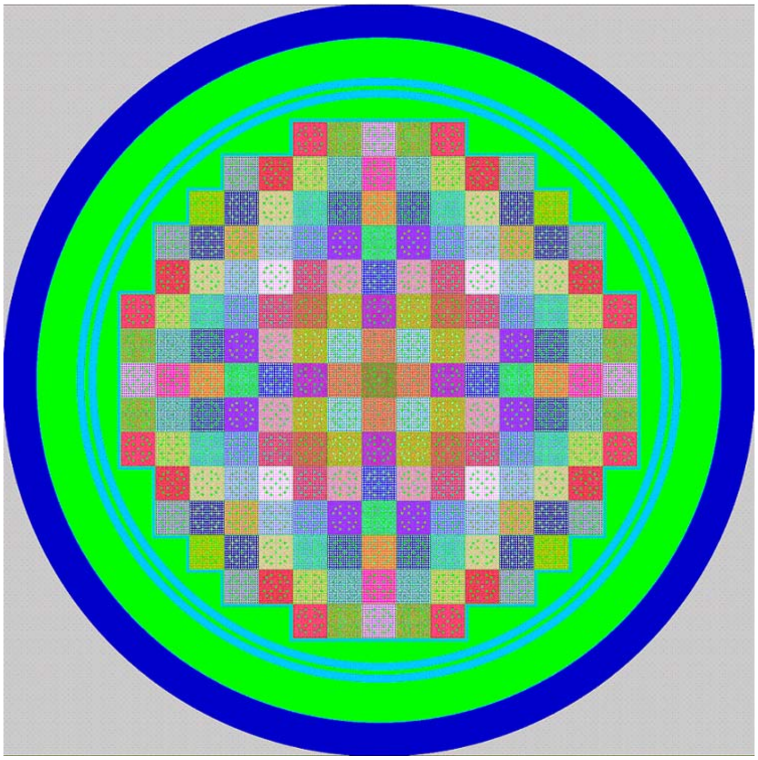


Figure 9. Commercial Reactor Critical (CRC) Model

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 and would be a follow-on 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; i.e., data that might be needed to address burnup credit for BWR SNF. Initial planning activities were initiated in 2005 with an anticipation for critical experiments to begin in 2007.

Through an NRC-supported agreement with Belgonucléaire, ORNL will also be able to assess critical experiments performed as part of the REBUS international program using the VENUS critical facility. These experiments involve critical UO_2 pin lattice configurations with portions of commercial BWR and PWR SNF assemblies inserted in the middle of the configuration. Final documentation of the critical experiment should be received by the end of 2005 and ORNL will initiate an evaluation of the experiment in 2006.

During 2005, ORNL staff also initiated interactions with officials from the Japan Atomic Energy Agency (JAEA), a unification of the former Japan Atomic Energy Research Institute and the Japan Nuclear Cycle Development Institute, relative to obtaining data on recently performed (in Japan) critical experiments that include select fission products. Preliminary experimental data has been received. Although these experiments were performed in support of dissolver (reprocessing) activities, they may have relevance to validation of burnup credit for transportation, and hence their applicability will be assessed. In addition, it is anticipated that through these interactions ORNL may have an opportunity to consult on the design of future fission product critical experiments in Japan (planned for 2007-2008) that are focused on spent fuel in storage and transport configurations.

4 DATA BASE OF ISOTOPIC ASSAY DATA FOR PWR FULL BURNUP CREDIT

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 that there is 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. Under this project, work has been ongoing to identify and assess potential sources of data that can support a strengthened technical basis for fission product credit.

Figure 10 illustrates the individual reactivity worth or importance of the major fission products for Westinghouse 17 × 17 SNF loaded in the GBC-32. The worth is expressed as the change in k_{eff} resulting from removing that fission product from the SNF material composition after a 5-y cooling time. The relative importance of each fission product depends on the burnup, enrichment, and (to a lesser extent) the cooling time of the fuel; however, the top six fission products (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{151}Sm, and ^{155}Gd) are unchanged and account for more than 75% of the total worth of the 15 fission products examined under nearly all conditions. These six fission products are the focus of this project's efforts to obtain and assess both destructive assay data and critical experiment data.

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. [16] Of the 56 PWR spent fuel samples that had been evaluated by ORNL prior to 2005 [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. [17] Table 5 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 5 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) [18] and the Japanese Takahama Unit 3 PWR fuel measurements performed by the Japan Atomic Energy Research Institute. [19]

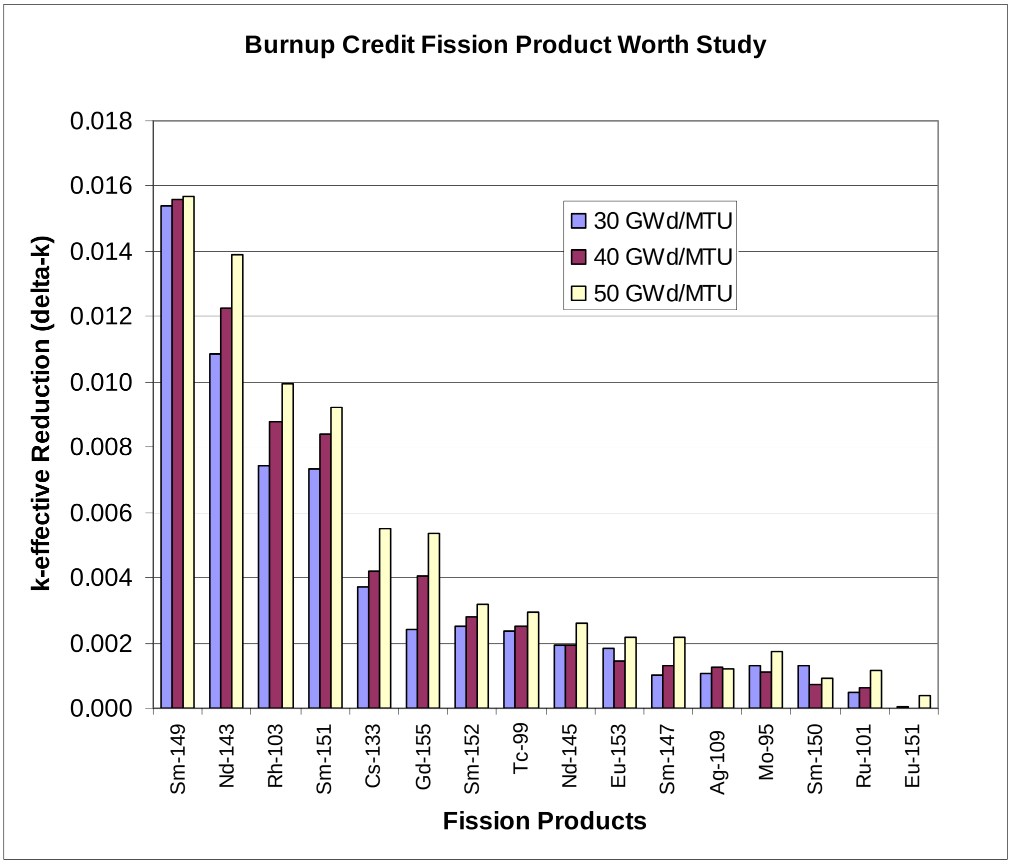


Figure 10. Fission Product Worth Calculated for WE 17 x 17 SNF Assemblies with 4 wt% Initial Enrichment and Loaded in the GBC-32 after a 5-y Cooling Time

Table 5. 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

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 is not currently being utilized. The calculated-to-experiment (C/E) ratio obtained for the measurements noted in Table 5 was 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.

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. [11] The data from these programs are proprietary but through the contract with Cogema (one of the optional purchases under the contract discussed in Section 3), 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 is 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 (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 (non reconstituted), and (4) the fact that the fuel is likely well characterized (because it was selected specifically to support code validation in France). Although not believed to be a significant issue, any differences between the operations of French plants as compared with domestic plants may introduce subtle biases in the measurements that may not be applicable to domestic plants. 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.

One difficulty which needs to be further addressed by ORNL in cooperation with EPRI is acquisition of the measured actinide data for the SNF samples. Different parties in France have the rights to the measured actinide data and so the actinide information is not obtainable through the existing ORNL contract with Cogema. Lack of actinide data to verify the fissile material depletion and plutonium production would seriously limit the ability to resolve any fission product discrepancies and would reduce the value of the data for code validation. These issues will be addressed in 2006.

Belgonucléaire 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 evaluating is small and the burnup range is high (> 50 GWd/MTU). The data will be commercial proprietary for a period of 3 years after the final report is issued, expected late in 2005. These data will be evaluated by ORNL in 2006 and included in a publicly distributed data base at the end of the 3-y proprietary period.

The Spanish Nuclear Safety Council, Consejo de Seguridad Nuclear (CSN), together with other Spanish and international partners, has conducted destructive assay measurements on eight high-burnup spent fuel samples. The measurements include all of the important burnup credit fission products. The high-burnup fuel rods selected for this program were obtained from reconstituted fuel assemblies. Because reconstitution is not typical in most commercial fuel assemblies, the value of the assay data for validation of commercial fuel has not been clearly established. During 2005, ORNL worked with CSN to analyze the assay measurements and obtained high, but similar C/E results as obtained in Spain. The reason for the discrepancy is unclear and ORNL is working with CSN and its partners to understand, and potentially resolve the issue. Plans for some independent measurements that will hopefully bring some resolution to the current interpretation issues are in place for 2006.

Sources of Additional Assay Data – Non-Proprietary

In 2005, ORNL contracted with PNNL to investigate and assess whether there are existing, US-origin spent nuclear fuel 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 the PNNL as part of the Approved Testing Material (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 making measurements on some or all of these samples in early 2006.

A major activity in the last half of 2005 has been work to re-assess reported measurements of Three Mile Island Unit 1 (TMI-1) SNF that were performed circa 1999 to support the YMP. [20] An earlier assessment of the TMI-1 data by ORNL deemed the TMI-1 data not suitable for use to obtain 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 Yucca Mountain Project [21] 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 Belgonucléaire programs. For example, Ref. 21 reports 30 – 40% differences between measured and calculated predictions for ^{239}Pu while re-analysis performed by ORNL in 2005 using state-of-the-art multidimensional reactor physics codes (both SCALE and HELIOS) show discrepancies of 10-20%. This compares to 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. [22] The reactor conditions experienced by these fuel samples are not well known and there are several suspected local conditions [22] that could significantly impact the predictions and are potentially the reason for the large C/E discrepancies.

Nevertheless, the difficulty with obtaining the quantity and quality of measured assay data for fission product nuclides has led ORNL to re-visit 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 reasons) might be similar to a small number of samples deemed to be of high quality (accurate radiochemical measurements with well-known reactor conditions). Thus, recently ORNL has 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 re-investigation, pending further work in 2006, is that the TMI-1 samples are not 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 as a result of adding 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.

5 NUCLEAR DATA ASSESSMENT, MEASUREMENT AND EVALUATION

Background and Approach

The technical rigor (physics measurements and evaluations to smoothly fit data over entire energy range) utilized in acquiring current fission product cross-section data is deficient 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 fission product nuclides (i.e., ^{103}Rh, ^{143}Nd, ^{149}Sm, ^{151}Sm, ^{133}Cs, and ^{155}Gd). This assessment, together with opportunities to leverage with cost-free international partners who are interested in improved cross-section data, will be used to prioritize and pursue improved cross-section measurements and evaluations. As needed and justified, new measurements in ORNL's Oak Ridge Electron Linear Accelerator (ORELA) facility will be performed and new evaluations and production cross-section libraries will be prepared that are consistent with the quality and rigor now provided in the actinide data. The fission product evaluations and subsequent production cross-section libraries developed under this activity will be distributed subsequent to testing and verification.

Assessment and Measurement Progress

During 2005, a draft report was prepared to document ORNL's assessment of the current US Evaluated Nuclear Data Files (ENDF) relative to cross section data for fission product nuclides. The report identifies recent improvements to the fission product data and identifies deficiencies that need to be addressed through new measurements or improved evaluation techniques. ORNL has also worked with the Institute of Reference Materials and Measurements (IRMM) in Belgium to develop a DOE-Euratom Action Sheet that will enable collaborations to improve fission product cross-section data. The draft assessment report when supplemented by input from potential international partners provides a plan for work in 2006 and beyond. Another ORNL report, published in 2005 [23], ranks the importance of fission product cross sections to the reactivity of spent nuclear fuel in a cask. Nuclides whose cross sections are important during irradiation (i.e., contribution to production of key nuclides important to reactivity) were identified and the cross section data for these nuclides will be assessed and evaluated in a priority consistent with their importance to burnup credit.

In order to support European nuclear applications, IRMM has interest in performing measurements and evaluations to improve cross-section data for the same fission products that are of interest to burnup credit for transport and storage applications. ORNL traveled to Belgium in September to work with IRMM staff on performing new cross-section measurements for ^{103}Rh. Currently ORNL staff are working with IRMM to assess the quality of the measured data and determine if perform additional measurements at ORELA are needed. A new evaluation for ^{103}Rh should be completed in 2006 with improved resonance data and improved uncertainty estimates that will enable utilization of the TSUNAMI-IP tool to propagate data uncertainties to the k_{eff} values. Another near-term focus of the ORNL-IRMM collaboration is completion of a new evaluation and corresponding covariance data for ^{133}Cs. The new evaluation will incorporate recent measurements made at IRMM in the thermal energy range. Also, ORNL has been working with IRMM to clarify the needs relative to fission product sample preparation (i.e., preparation of new samples, exchange of existing samples at ORNL or IRMM, etc.) for future measurements.

6 OTHER ACTIVITIES

Data for Improved Safety Analyses

ORNL utilized 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 looking to provide a technical basis for recommending bounding assumption values that can be used in the safety analysis. A reduction in conservative values recommended in earlier reports is anticipated and 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 will continue into 2006.

Automated Loading Curve Software

To facilitate work under this project and to provide NRC staff with an easy-to-use confirmatory tool, ORNL updated its SCALE graphical user interface to enable use with the STARBUCS sequence – a sequence that automates the SNF isotopic prediction and the criticality analysis into one calculation. Limited effort was spent on a prototypic version of STARBUCS that automatically generates burnup credit loading curves. Efforts to develop a formal automated loading curve sequence for the SCALE code package is expected to begin in 2006.

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. Approaches that are simple, but reliable, for using burnup credit to assure full cask loadings of all inventory up to 5 wt% will also be explored.

7 SUMMARY

This report has summarized the activities performed by this project during 2005. 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 $150 – 400M.

The highlight of the year was the successful negotiation of a contract with Cogema to gain access to the results of the extensive experimental program conducted in France in support of burnup credit. The highest priority data has been obtained (HTC critical experiment set in final form and the PF or fission product critical experiment set in draft form) and is 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 continues to be a challenge. ORNL has investigated all known sources of assay data and initiated a new effort to re-assess and provide guidelines on utilizing the TMI-1 measured data which provides 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) assure a solid technical basis consistent with cost and benefit is established. Thus, the CRC data continues to be assessed for applicability to cask systems, efforts to improve the cross-section data for fission product nuclides have been initiated, and activities to increase the data base via domestic (e.g., new critical experiments at SNL and assay data measurements at PNNL) or international (e.g., participation in international research programs) are ongoing. By the end of 2006, ORNL is expected to be in a position to provide draft recommendations on implementing fission product credit using the data that has been obtained and demonstrate where future work (e.g., planned experimental data or an improved reactor operating history data base) might improve implementation of full burnup credit.

8 REFERENCES

1. Spent Fuel Project Office, Interim Staff Guidance – 8, Revision 2, "Burnup Credit in the Criticality Safety Analysis of PWR Spent Fuel in Transport and Storage Casks," September 27, 2002.

2. J.C. Wagner and D.E. Mueller, "Updated Evaluation of Burnup Credit for Accommodating PWR Spent Nuclear Fuel to High-Capacity Cask Designs," Presented at the 2005 Topical Meeting on Nuclear Criticality Safety, Knoxville, TN, September 19-22, 2005.

3. *RW-859 Nuclear Fuel Data*, Energy Information Administration, U.S. Department of Energy (Oct. 2004).

4. J. C. Wagner, *Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit*, NUREG/CR-6747 (ORNL/TM-2000/306), U. S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, Oak Ridge, Tenn., October 2001.

5. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 5, Vols. I-III, April 2005.

6. I. C. Gauld, *Strategies for Application of Isotopic Uncertainties in Burnup Credit*, NUREG/CR-6811 (ORNL/TM-2001/257), U.S. Nuclear Regulatory Commission (June 2003).

7. D. G. Dippold, "Burnup Credit in Nuclear Waste Transport: An Economic Analysis", p. 125 – 146, *Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks, February 21-22, 1988.* SAND89-0018 (October 1989).

8. *International Handbook of Evaluated Criticality Safety Experiments*, NEA/NSC/DOC(95)03, September 2005 Ed., NEA/OECD, September 2005.

9. B. L. Broadhead, B. T. Rearden, C. M. Hopper, J. J. Wagschal, and C. V. Parks, "Sensitivity and Uncertainty-Based Criticality Safety Validation Techniques," *Nucl. Sci. Eng.* **146**, 340–366 (2004).

10. B. T. Rearden, "Perturbation Theory Eigenvalue Sensitivity Analysis with Monte Carlo Techniques," *Nucl. Sci. Eng* . **146**, *367–382 (2004).*

11. C. Lavarenne, E. Letang, O. Denkens, M. Doucet, J.P. Grouiller, N. Thiollay, E. Guillou, Taking Burnup Credit Into Account in Criticality Studies: The Situation As it is Now and the Prospects for the Future, International Conference on Nuclear Criticality Safety (ICNC'99), Versailles, September 1999.

12. D. E. Mueller and G. A. Harms, "Using the SCALE 5 TSUNAMI-3D Sequence in Critical Experiment Design," Trans. Am. Nucl. Soc. 93, 263-266 (2005).

13. CRWMS M&O 1998 Summary Report of Commercial Reactor Criticality Data for Crystal River Unit 3, B00000000-01717-5705-00060 REV 01, Las Vegas, NV: CRWMS M&O. MOL.19980728.0189.

14. R. A. Kochendarfer, J. M. Scaglione, and D. A. Thomas, *CRC Reactivity Calculations for Crystal River Unit 3*, B00000000-01717-0210-00002-00 (C), MOL.19980728.0004.

15. Monte Carlo Team, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5", LA-UR-03-1987, Los Alamos National Laboratory, April 2003.

16. C. V. Parks, M. D. DeHart, and J. C. Wagner, *Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel*, NUREG/CR-6665 (ORNL/TM-1999/303), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2002.

17. M. C. Brady-Rapp and R. J. Talbert, *Compilation of Radiochemical Analyses of Spent Nuclear Fuel Samples*, PNNL-13677, Pacific Northwest National Laboratory, September 2001.

18. M. C. Brady-Raap, Compilation of Radiochemical Analyses of Spent Nuclear Fuel Samples, Pacific Northwest National Laboratory, PNNL-13677, September 2001.

19. Y. Nakahara, K. Suyama, and T. Suzaki, *Technical Development on Burnup Credit for Spent LWR Fuels*, JAERI-Tech 2000-071, Japan Atomic Energy Research Institute, Tokai Research Institute, October 2000. Report translated as ORNL/TR-2001/01, Oak Ridge National Laboratory, January 2002.

20. Bechtel SAIC Company, *Three Mile Island Unit 1 Radiochemical Assay Comparisons to SAS2H Calculations*, CAL-UDC-NU-000011, Rev. A, Office of Civilian Radioactive Waste Management, Las Vegas, Nev., 2002.

21. J. M. Scaglione, "Isotopic Bias and Uncertainty for Burnup Credit Applications," presented at the American Nuclear Society 2002 Winter Meeting, Washington, D.C., November 17–21, 2002.

22. TMI-1 Cycle 10 Fuel Rod Failures – Volume 1: Root Cause Failure Evaluations, EPRI

Report TR-108784-V1, 1998.

23. I. C. Gauld and D. E. Mueller, *Evaluation of Cross-Section Sensitivities in Computing Burnup Credit Fission Product Concentrations,* ORNL/TM-2005/48, Oak Ridge National Laboratory, Oak Ridge, TN, August 2005.

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