IAEA Conference Paper
Thursday, February 16, 2006

Nuclear Science and Technology Division

A Coordinated U.S. Program to Address Full Burnup Credit in Transport and Storage Casks

C. V. Parks and J. C. Wagner

Oak Ridge National Laboratory*
PO Box 2008
Oak Ridge, TN 37831-6170 USA
(865) 574-5280
parkscv@ornl.gov

Presented at
The International Atomic Energy Agency (IAEA) Technical Committee Meeting on Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition,
August 29 – September 2, 2005,
London, UK

*Managed by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy.

A coordinated U.S. program to address full burnup credit in transport and storage casks

C.V. Parks, J.C. Wagner
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, Tennessee 37831-6170
USA

Abstract. The benefits of burnup credit and the technical issues associated with utilizing burnup credit in spent nuclear fuel (SNF) casks have been studied in the United States for almost two decades. The issuance of the U.S. Nuclear Regulatory Commission (NRC) staff guidance for actinide-only burnup credit in 2002 was a significant step toward providing a regulatory framework for using burnup credit in transport casks. However, adherence to the current regulatory guidance (e.g., limit credit to actinides) enables only about 30% of the existing pressurized-water-reactor (PWR) SNF inventory to be transported in high-capacity (e.g., 32-assembly) casks. Work has been done to demonstrate that the allowable inventory percentage could potentially increase to nearly 90% if credit for fission products were allowed. Thus, Oak Ridge National Laboratory has worked with the U.S. Department of Energy Office of National Transportation (DOE/ONT), the NRC, and the Electric Power Research Institute (EPRI) to coordinate a research program that will (a) obtain and evaluate experiment data to support the safety basis for fission product credit validation, (b) investigate unresolved technical issues associated with PWR full burnup credit, and (c) recommend approaches for boiling-water reactor (BWR) burnup credit in transport and storage casks. This paper will review the program of research and discuss the progress to date.

1. Introduction

Safe, efficient, and effective management of spent nuclear fuel (SNF) from U.S. commercial nuclear power plants will demand increasing attention to transport and storage in casks. Historically, 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-year cooling time for transport of domestic 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 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 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 (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 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 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.

2. Data base of critical experiments for full burnup credit

2.1. Background and approach

The potential benefits of burnup credit relative to the increased inventory of PWR SNF that could be transported in high-density casks have been demonstrated in Ref. [2]. The cost savings from this inventory increase varies from a minimum of $156M to $400M depending on the assumptions relative to cask sizes. The project being discussed in this paper is seeking 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 the ISG-8R2 to actinide-only is based largely on the lack of 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:

(a) critical experiments within the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (*IHECSBE*) [3];
(b) proprietary critical experiment data;
(c) commercial reactor criticals (CRCs); critical state points from operating reactors; and
(d) 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 [4] and incorporated within Version 5 of the SCALE code system [5]. The TSUNAMI-3D sequence within SCALE uses first-order linear perturbation theory [6] 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 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 [4] 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 [7] loaded with Westinghouse 17×17 fuel (see Fig. 1) 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.

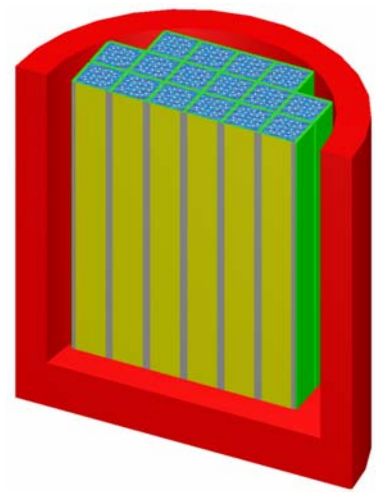

FIG. 1. GBC-32 cask model.

2.2. Assessment of IHECSBE and French proprietary experiments

As part of this project, ORNL was able to negotiate a multioption 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 [8] 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 uranium and plutonium isotopic compositions similar to U(4.5%)O_2 fuel with a burnup of 37,500 MWd/MTU. The experimental series, referred to as the HTC experiments, investigated 156 configurations divided into 4 groups, as illustrated in Fig. 2. The first group is a single clean-water-moderated and water-reflected array of HTC rods with the pin pitch varied from 1.3 to 2.3 cm. The second group is similar to the first, except that boron or gadolinium is dissolved in the water 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.

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. (Actinides and fission products are included in the reference model.) 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 ≥ 0.8, with none having c_k values ≥ 0.9. (Additional non-HTC MOX experiments continue to be assessed.) However, the strong applicability of the HTC MOX experiments is demonstrated by the fact that 152 of the 156 configurations have c_k values ≥ 0.8, with 143 c_k values ≥ 0.9. The few experiments with c_k < 0.9 all had high soluble gadolinium concentrations to simulate systems in fuel reprocessing. The results of these studies confirm the significant value of the HTC experiments for criticality validation of the primary actinides and the weaker validation basis that exists without the HTC experiments.

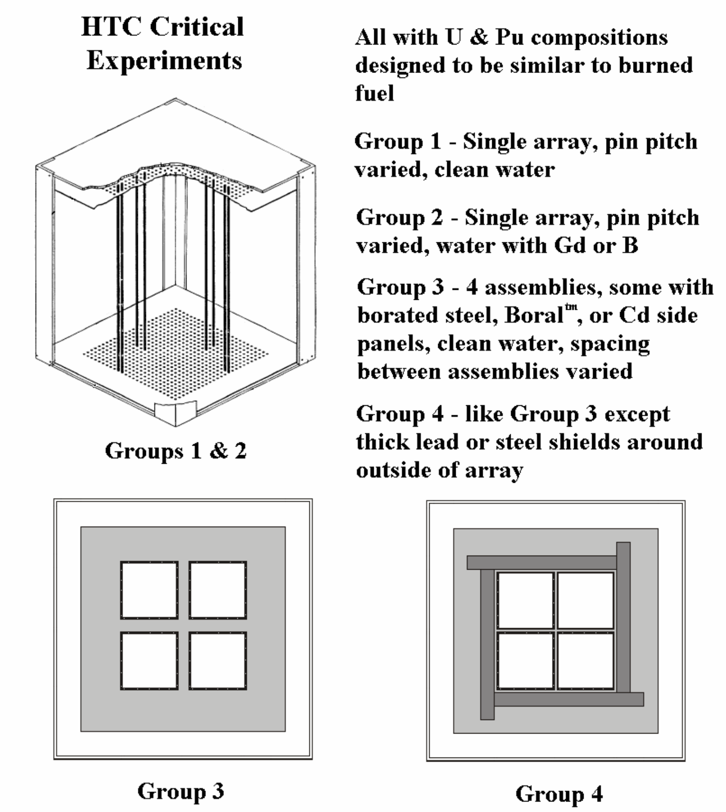

FIG. 2. French HTC critical experiments.

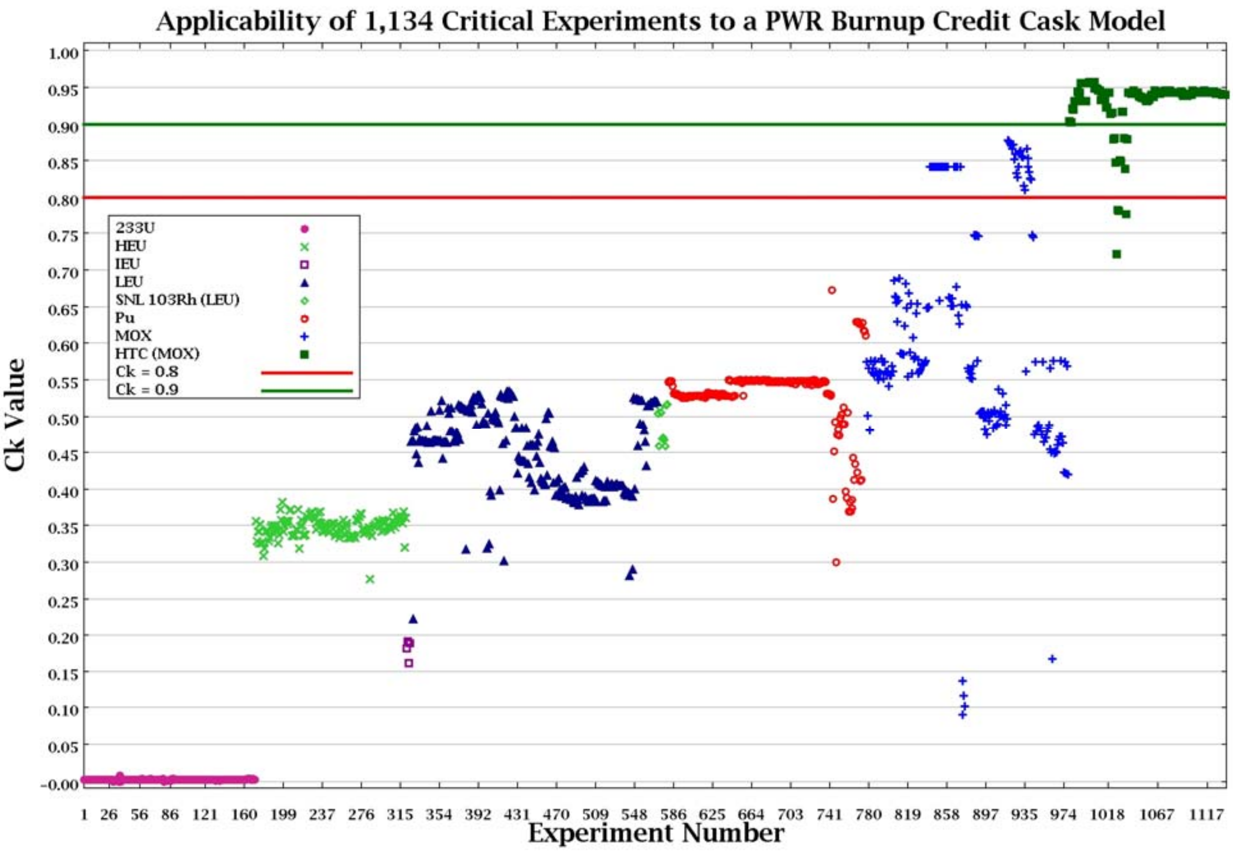

FIG. 3. Critical experiment applicability to burnup credit.

However, the HTC experiments do not provide validation for the fission product compositions in the SNF, and work has been initiated to assess critical experiments that will address this validation need. 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.

The 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 (see Sect. 2.4) to ensure maximum applicability [9].

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 uses a central tank filled with water, borated water, or fission product solution. The central tank is surrounded by $U(4.7)O_2$ fuel rods in water. The second group of experiments uses a central tank containing an 11 × 11 array of either $U(4.7)O_2$ or HTC MOX rods in uranyl nitrate solutions with dissolved fission products. The central tank is surrounded by $U(4.7)O_2$ fuel rods in water. The third group of experiments uses a large tank containing an array of either $U(4.7)O_2$ 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 4 of the 147 critical configurations are sufficiently similar to the GBC-32 cask model to yield c_k values greater than 0.8. These four configurations are nearly identical and yield c_k values of about 0.97. Preliminary observations indicate that the HTC MOX rods dominate the overall c_k comparison between these experiments and the GBC-32 model. Work in progress involves investigation of the sensitivity profiles by nuclide. Using TSUNAMI-IP, the goal of the project is to quantify an uncertainty allowance for the fission products by using the sensitivity profile information for all the criticals and the limited number of applicable critical configurations that have high c_k values.

2.3. Assessment of commercial reactor critical (CRC) configurations

Work currently in progress includes modeling and 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 that are documented in great detail in the Yucca Mountain Project (YMP) reports [10–11]. The CRC state points require very large, complex computational models with the following information needed for completeness: 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. Figure 4 shows an overhead view of the Crystal River Unit 3 model as generated by the SCALE graphical display package.

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, 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.

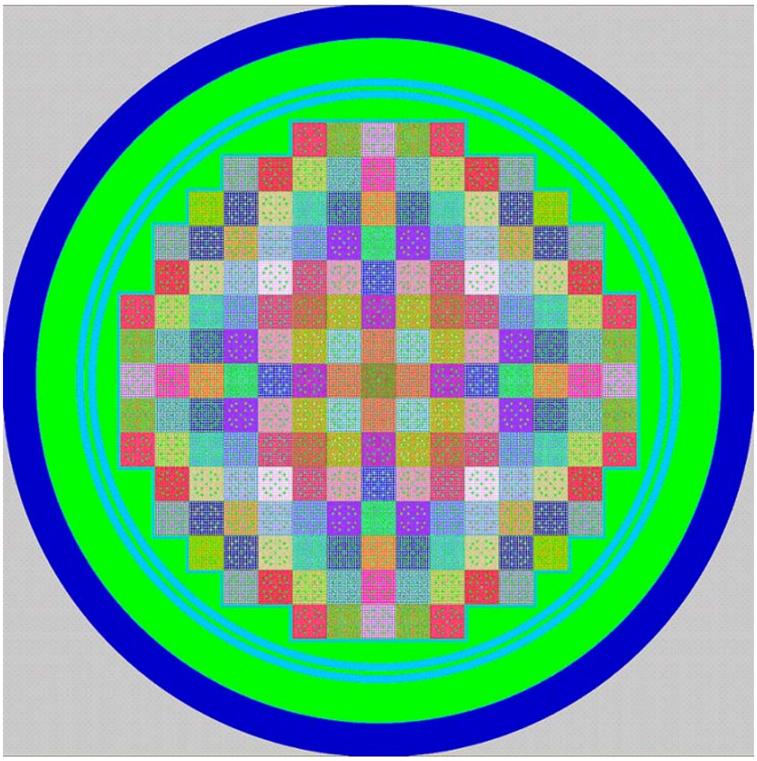

FIG. 4. Commercial Reactor Critical (CRC) model.

2.4. Proposed new critical experiments

This coordinated 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 (e.g., data that might be needed to address burnup credit for BWR SNF). Planning activities were initiated in 2005.

Through an NRC-supported agreement with Belgonucleaire, 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.

3. Data base of isotopic assay data for PWR full burnup credit

3.1. 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 indicates 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. Figure 5 illustrates the individual reactivity worth or importance of the major fission products for Westinghouse 17 × 17 SNF loaded in the GBC-32. 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.

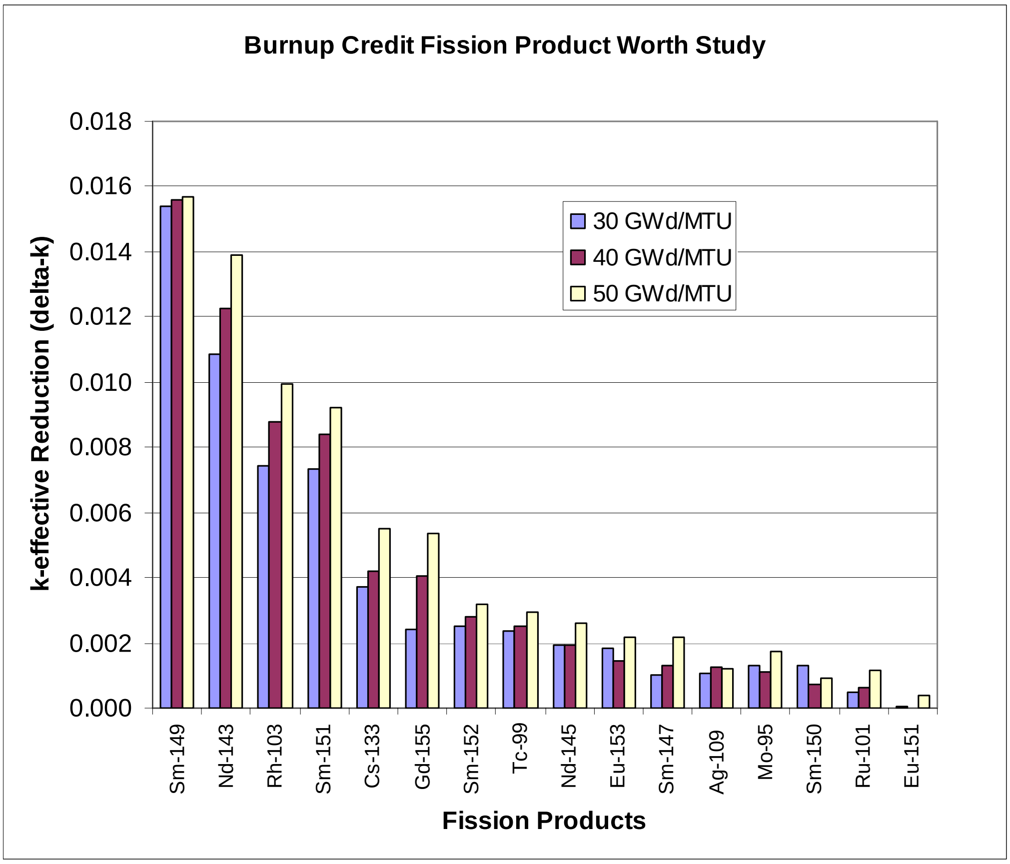

FIG. 5. Fission product worth calculated for WE 17 × 17 SNF assemblies with 4 wt % initial enrichment and loaded in the GBC-32 after a 5-year cooling time.

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) [12]. Of the 56 PWR spent fuel samples that had been evaluated by ORNL prior to 2005 [13], 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 [14]. Table I 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 I 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. Klopin Radium Institute (St. Petersburg, Russia) [15] and the Japanese Takahama Unit 3 PWR fuel measurements performed by the Japan Atomic Energy Research Institute [16].

Table I. 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 (a) collecting all of the relevant data into a single data base and (b) identifying measurement data that are not currently being utilized. The calculated-to-experiment (C/E) ratio obtained for the measurements noted in Table I 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.

3.2. 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 spent fuel assay measurements in support of fuel inventory and fuel cycle studies, including burnup credit [8]. The data from these programs are proprietary, but through the contract with COGEMA (one of the optional purchases under the contract discussed in Sect. 2), 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 (a) the high-accuracy radiochemical analysis methods employed, (b) the wide range of enrichments and burnups (covering most commercial U.S. fuels), (c) the use of standard commercial fuel assemblies (nonreconstituted), and (d) the fact that the fuel is likely well characterized (because it was selected specifically to support code validation in France). Although not thought 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.

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, 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.

3.3. Sources of additional assay data—nonproprietary

In 2005, ORNL contracted with PNNL to investigate and assess whether there are existing, U.S.-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 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 in the last half of 2005 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 [17]. 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 [18] 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. [19] reports differences of 30–40% between measured and calculated predictions for ^{239}Pu. Reanalysis 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 with typical calculated-to-measured differences of ±5% for ^{239}Pu. The TMI-1 fuel was originally selected for postirradiation examination because it had experienced extreme crud buildup during irradiation and possible fuel cladding failure of the assembly [19]. The reactor conditions experienced by these fuel samples are not well known. Several suspected local conditions [19] that could significantly impact the predictions are potentially the reason for the large 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 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, 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 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 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.

4. 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 its 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.

5. Other activities

5.1. 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 working 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 is a continuing effort.

5.2. 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.

6. Summary

This report has summarized the current U.S. project on burnup credit and the activities performed to date. The highest-priority data have been obtained (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 (a) provide flexibility in future safety analyses and (b) 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 (e.g., new critical experiments at SNL and assay data measurements at PNNL) or international participation in research programs. By the end of 2006, ORNL is seeking 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 reactor operating history data base) might improve implementation of full burnup credit.

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] WAGNER, J.C., MUELLER, D.E., "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] *International Handbook of Evaluated Criticality Safety Experiments*, NEA/NSC/DOC(95)03, September 2005 Ed., NEA/OECD, September 2005.

[4] BROADHEAD, B.L., REARDEN, B.T., HOPPER, C.M., WAGSCHAL, J.J., PARKS, C.V., "Sensitivity and Uncertainty-Based Criticality Safety Validation Techniques," *Nucl. Sci. Eng.* **146**, 340–366 (2004).

[5] *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, ORNL/TM-2005/39, Version 5, Vols. I–III, April 2005. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.

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

[7] WAGNER, J.C., *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, October 2001.

[8] LAVARENNE, C., LETANG, E., DENKENS, O., DOUCET, M., GROUILLER, J.P., THIOLLAY, N., GUILLOU, E., "Taking Burnup Credit Into Account in Criticality Studies: The Situation As It Is Now and the Prospects for the Future," in *Proceedings of International Conference on Nuclear Criticality Safety (ICNC'99)*, Versailles, September 1999.

[9] MUELLER, D.E., HARMS, G.A., "Using the SCALE 5 TSUNAMI-3D Sequence in Critical Experiment Design," *Trans. Am. Nucl. Soc*. **93**, 263–266 (2005).

[10] *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.

[11] KOCHENDARFER, R.A., SCAGLIONE, J.M., THOMAS, D.A., *CRC Reactivity Calculations for Crystal River Unit 3,* B00000000-01717-0210-00002-00 (C), MOL.19980728.0004.

[12] PARKS, C.V., DEHART, M.D., WAGNER, J.C., *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.

[13] GAULD, I.C., *Strategies for Application of Isotopic Uncertainties in Burnup Credit*, NUREG/CR-6811 (ORNL/TM-2001/257), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, June 2003.

[14] BRADY-RAPP, M.C., TALBERT, R.J., *Compilation of Radiochemical Analyses of Spent Nuclear Fuel Samples*, PNNL-13677, Pacific Northwest National Laboratory, September 2001.
[15] BRADY-RAAP, M.C., *Compilation of Radiochemical Analyses of Spent Nuclear Fuel Samples*, Pacific Northwest National Laboratory, PNNL-13677, September 2001.
[16] NAKAHARA, Y., SUYAMA, K., SUZAKI, T., *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.
[17] 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, NV, 2002.
[18] SCAGLIONE, J.M., "Isotopic Bias and Uncertainty for Burnup Credit Applications," presented at the *American Nuclear Society 2002 Winter Meeting*, Washington, D.C., November 17–21, 2002.
[19] *TMI-1 Cycle 10 Fuel Rod Failures — Volume 1: Root Cause Failure Evaluations*, EPRI Report TR-108784-V1, 1998.