REGULATORY PERSPECTIVE ON COMPUTER CODE VALIDATION FOR BURNUP CREDIT CRITICALITY ANALYSES FOR SPENT NUCLEAR FUEL TRANSPORTATION PACKAGES

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ABSTRACT

In an effort to increase the capacity of the spent fuel transportation packages, applicants for certificates of compliance under 10 CFR Part 71 (Ref. 1) have increasingly sought credit for the reduction in reactivity because of fuel burnup, or burnup credit, in their criticality safety analyses. NRC's Division of Spent Fuel Storage and Transportation published Interim Staff Guidance 8 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," in May of 1999 (Ref. 2), with subsequent revisions in July of 1999 (Ref. 3) and September of 2002 (Ref. 4). This document provides guidance regarding acceptable approaches to burnup credit criticality analyses for intact spent PWR assemblies in transportation packages.

New data and approaches have been developed since the publication of ISG-8 Revision 2. The NRC is now considering revising ISG-8 based on these new data and methodologies for burnup credit. This paper will detail the technical basis supporting the current recommendations in ISG-8 regarding acceptable validation techniques for isotopic depletion and criticality codes necessary for burnup credit criticality analyses. Additionally, this paper will discuss staff considerations for a potential revision to ISG-8, including new data and computational techniques, which have recently become available for isotopic depletion and criticality code validation.

I. INTRODUCTION

Transportation of spent nuclear fuel (spent fuel) is governed by the regulations set forth in Part 71 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 71), "Packaging and Transportation of Radioactive Material." The requirements applicable to criticality safety include 10 CFR 71.55(b), which requires that the most reactive credible configuration of the contents, moderation to the most credible extent (with fresh water) and optimum reflection be considered in analyses for demonstrating criticality safety. In addition, 10 CFR 71.83 requires that the packaging of fissile materials for which pertinent properties (e.g., isotopic abundance, mass, concentration, and degree of irradiation) are unknown shall be done assuming these properties have "credible values that will cause the maximum neutron multiplication"

Compliance with these requirements is typically demonstrated by computational methods. These analyses can be very detailed and complex, covering a wide range of conditions. Industry standards have been established to guide the use of these methods, including computational method validation and the establishment of biases and uncertainties. For example, these standards state that method validation and bias determination require the computational method be benchmarked against appropriate critical experiments (Ref. 5, 6, and 7).

Since the isotopic composition of spent fuel can vary significantly depending on the designs and irradiation histories of fuel assemblies, spent fuel transportation package vendors have traditionally evaluated criticality safety assuming the fuel is unirradiated (i.e., fresh). Because the isotopic composition of fresh fuel is well-specified, package designs based upon the bounding assumption of fresh fuel contents do not need to address the analytical complexities associated with spent fuel contents.

The fresh-fuel approach has typically been adequate, as the capacity of spent fuel transportation package designs submitted to the NRC for certification has been limited by other design parameters (e.g., heat load and radiation source terms). In these packages, subcriticality was maintained through use of spacing and/or flux traps between assemblies. However, more recent transportation package designs have sought to increase cask capacities in order to reduce the number of shipments and overall cost of spent fuel transportation. Also, initial enrichments have increased due to higher enrichment fuel now being used in reactors. Additionally, licensees are seeking to transport spent fuel in dry cask storage systems which have not been able to demonstrate compliance with the 10 CFR Part 71 criticality safety requirements with the fresh-fuel assumption. Therefore, spent fuel transportation package designs have recently sought credit for the reduction of reactivity in the spent fuel.

NRC's Division of Spent Fuel Storage and Transportation (SFST) developed Interim Staff Guidance 8 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," (Ref. 4) to provide guidance to the NRC staff for reviewing the use of burnup credit in criticality analyses for spent fuel transportation package designs. The initial ISG-8 (Ref. 2) was published in May 1999 and has been revised twice (July 1999 (Ref. 3) and September 2002 (Ref. 4)) as additional data has become available through various research programs. At each revision, the guidance has been modified to expand the burnup credit that may be applied in criticality analyses for spent fuel transportation packages, as can be supported by the available data. To date, burnup credit has not been considered for transportation of BWR spent fuel.

II. TECHNICAL BASIS FOR CODE VALIDATION IN ISG-8, REVISION 2

The spent fuel cask typically consists of three major parts: spent fuel contents, fuel basket, and cask body. The fuel basket is typically made of stainless steel structural materials and poison plates. The cask body is normally made of structural and shielding materials such as stainless steel, lead, aluminum, carbon steel and neutron shielding layers (typically borated resin). The material compositions of spent fuel are very complicated, including unburned ^{235}U and ^{238}U, actinides and fission products as a result of fuel depletion and uranium transmutation. To determine the bias and uncertainties of the criticality calculations for burnup credit casks, the computer codes used for cask criticality safety evaluations must be benchmarked to critical benchmark experiments with high similarities with the burnup credit casks in both material compositions and geometrical configurations in order to obtain reliable criticality safety evaluation results.

The burnup credit design analyses are mainly composed of depletion analyses and criticality analyses. The depletion analyses, which simulate nuclear fuel irradiation in reactor cores, produce the isotopic inventory that is used as input into criticality safety analyses for spent fuel assemblies inside transportation packages or storage casks. There are more than 1,500 different isotopes present in spent fuel; however, only a handful of isotopes are important with respect to criticality safety. They include most of the actinides and some of the fission products. The current version of ISG-8 (Revision 2) recommends that credit for burnup in transportation of spent fuel should be limited to the major actinide isotopes in spent PWR fuel (^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am). This is because there were data sufficient only for these isotopes to perform isotopic and criticality computer code validations. The following sections provide a summary of the isotopic and criticality code validation guidance under ISG-8 (Revision 2) and the supporting data.

II.A Benchmarking Data for Isotopic Depletion Computer Codes

As indicated earlier, the burnup credit recommended in ISG-8, Rev. 2 was limited to PWR actinide isotopes. In addition, the burnup credit was limited to a burnup of 50 GWd/MTU because the assay data were not available to support development of a bias and uncertainty beyond this burnup without unwarranted extrapolation. Table 1, shows the sources of data for ISG 8, Rev. 2 (Ref. 8).

The Trino Vercelles Power Plant is an 825-MW PWR located in Italy and operated by the Ente Nazionale per l'Energia Elettrica (ENEL). The steam generation plant and fuel were designed by Westinghouse Electric Corp. The reactor is based on one of the earlier Westinghouse designs and is different from most of the PWR designs in the United States. However, its design is similar to that of the U.S. Yankee Rowe PWR. Radiochemical assay data obtained from the analysis of three fuel assemblies were used as part of the basis for ISG 8, Rev.2. Two of the assemblies, numbered 509-104 and 509-032, were irradiated in the core during the first cycle only. The remaining assembly, identified as 509-069, was irradiated during both the first and second cycle.

The Turkey Point Unit 3 PWR, operated by Florida Power and Light Co., was designed by Westinghouse Electric Corp. The fuel assembly design is based on a 15 X 15 square lattice, with 21 positions containing control rod and instrumentation guide tubes. Radiochemical assay data obtained from the analysis of two fuel assemblies, identified as D01 and D04, were used as part of the basis for ISG-8, Rev. 2.

Obrigheim is a Siemens designed PWR which uses a 14 X 14 fuel assembly lattice. All the fuel assemblies analyzed were loaded in Obrigheim during cycles 3, 4 and 6. The samples were taken from six different spent fuel assemblies which were cut in half lengthwise and separately dissolved. Therefore, the Obrigheim data provides for benchmarking depletion computer codes with respect to predicting the average isotopic inventory in an assembly.

The H. B. Robinson Unit 2 fuel assembly design is a PWR 15 X 15 Westinghouse design. The Materials Characterization Center (MCC) at Pacific Northwest Laboratory (PNL) selected one assembly from H.B. Robinson-2 for the Approved Testing Material (ATM) program that was designed to provide a source of well-characterized spent fuel. The assembly selected (B05) for characterization had twelve burnable poison rods inserted during Cycle 1 and 2 and withdrawn during Cycle 3 and 4. The assembly had eight guide tubes and one instrument tube. The samples were taken from a spent fuel rod (N-9) that was adjacent to a burnable poison rod and diagonally next to a water-filled guide tube. Four samples were taken from the N-9 rod at various axial heights.

The Calvert Cliffs Unit 1 PWR uses Combustion Engineering (CE) fuel assemblies with a 14 X 14 pin lattice. Three spent fuel assemblies (D047, D101, and BT03) from Calvert Cliffs Unit 1 were also selected for characterization under the ATM program at PNL Ref., 8). D047 was loaded in Calvert Cliffs Unit 1 during cycles 2 through 5, D101 cycles 2 through 4, and BT03 cycles 1 through 4. A single rod was selected from each assembly for characterizations. The selected rods included rods next to the center water-filled guide tubes and from the periphery of the assemblies. Three samples were selected from each rod along the active fuel length.

The isotopic data from Takahama-3 (Ref. 9) became available after the issuance of ISG-8, Rev.2. The Takahama-3 reactor is a PWR with a rated thermal power of 2652 MW and operates with a 17 X 17 fuel assembly design. Radiochemical isotopic assay measurements were performed on two different Takahama-3 fuel assemblies: NT3G23 and NT3G24. Both fuel assemblies feature 14 integral burnable gadolinia-bearing (Gd2O3) fuel rods containing 2.6 wt% U-235 and 6.0% gadolinia while the standard fuel rods contain 4.11 wt% U-235 enrichment. The assemblies also had 25 water-filled guide tubes.

II.B Benchmarking Data for Criticality Computer Codes

For validation of burnup credit cask criticality calculations, several sources of critical experiments were publicly available for the major actinides at the time ISG-8, Rev. 2, was published. Although not entirely representative of commercial spent fuel in a cask-like environment, there were some fresh UO_2 and MOX critical experiments available in the public literature that can be used for criticality code validation for analyses involving the eight major actinides. No such experiments were publicly available at that time for fission products. The additional negative reactivity provided by fission products was therefore maintained as a margin to offset potential uncertainties due to unanticipated fuel operating conditions that increase reactivity. This margin would also cover potential uncertainties associated with the dissimilarity between UO_2 and MOX critical experiments used for actinide criticality validation, and the spent fuel cask systems being evaluated. The following are the sources for critical benchmark experiments used in the development of ISG-8, Rev. 2.

The topical report for the DOE-RW (Ref. 10) program provides a collection of critical benchmarks that includes both low enriched UO_2 (180) and MOX (47) applications. The MOX fuel critical experiments use a wide range of ^{239}Pu-to-^{241}Pu ratios to simulate different burnups. The DOE Fissile Material Disposition Program (FMDP) provides additional critical experiments (Ref. 11). The purpose of these experiments is the validation of a MOX transport package for that program. A number of the 29 MOX systems included in the FMDP are also included in the DOE-RW topical report. A third source was a collection of some 425 experiments that was developed for sensitivity/uncertainty studies (Ref. 12). Of the experiments in this collection, 168 were low enriched UO_2 systems and 76 were MOX systems. Some of these experiments are also included in the collections of experiments for the two DOE programs previously mentioned. The remaining source is a collection of experiments that were designed for two separate studies (Ref. 13), a study for transport of HEU in DOE-specific applications and a study in the validation of ^{241}Am in burnup credit. Of the 235 experiments in this last source, 113 were low enriched UO_2 systems and 7 were MOX systems. The MOX systems were taken from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (IHECSBE) (1998 edition) (Ref. 15)and were those that included ^{241}Am in their specification.

Further, ORNL (Ref. 14) also evaluated 12 state points from a collection of Commercial Reactor Criticals (CRCs). Seven state points are from PWR state points with remainder BWR state points. Six of these state points, taken from three reactors, were considered in work supporting ISG-8, Rev. 2. One state point is a Beginning of Cycle (BOC) Hot Zero Power (HZP) configuration from TMI-1 with a core-average burnup of 11 GWd/MTU and an initial enrichment range of 2.6 to 2.9 wt %. One BOC HZP case and one End of Cycle (EOC) Hot Full Power (HFP) case were taken from Surry-1 with a core-average burnup of 7 and 14 GWd/MTU, respectively, and an initial enrichment range of 1.9 to 3.3 wt %. The remaining three state points were taken from the Sequoyah reactor. These include one BOC HZP, one BOC HFP and one Middle of Cycle (MOC) case. The core-average burnup was 11 (both BOC cases) and 19.2 GWd/MTU, respectively, and the range of initial enrichment was 2.6 to 3.8 wt %.

III. CONSIDERATIONS FOR A POTENTIAL ISG-8, REVISION 3 REGARDING CODE VALIDATION

Given that ISG-8, Rev. 2, was published almost 7 years ago and it was focused on credit for actinide isotopes only, NRC is now evaluating the ISG for potential revision to consider credit beyond actinide isotopes as well as new data and calculation techniques that have recently become available

III.A. Isotopic Depletion Analysis Benchmarking

As discussed earlier, the benchmarking data for isotopic validation under ISG-8, Rev. 2 were focused on the actinide isotopes only. For ISG-8, Rev.3, NRC staff will focus on adding newly available actinide data, as well as fission product isotopic data from high burnup spent nuclear fuel. The new data will include samples with burnups as high as 70 GWd/MTU and initial enrichments as high as 4.7 wt%. The sources of these data will include samples from Three Mile Island Unit 1 (TMI -1), as well as the MOX and UOX LWR Fuels Irradiated to High Burnup (MALIBU), Actinides Research In A Nuclear Element (ARIANE), and Reactivity Tests for a Direct Evaluation of the Burnup Credit on Selected Irradiated LWR Fuel Bundles (REBUS) programs.

Measurements of 19 spent fuel samples from the TMI-1 reactor have been performed with the support of the U.S. Department of Energy Yucca Mountain Project. Fuel rods were obtained from two separate assemblies, identified as NJ05YU and NJ070G. Radiochemical analyses were performed at two independent experimental facilities: Argonne National Laboratory (ANL) and the General Electric–Vallecitos Nuclear Center (GE-VNC). Measurements on 11 of the TMI-1 samples from rod H6 of assembly NJ05YU were performed in 1998 and 2000 at ANL. The other eight TMI-1 samples, from rods O1, O12, and O13 of assembly NJ070G, were analyzed in 1999 at GE-VNC. Fuel rod H6 had an initial enrichment of 4.013 wt% ^{235}U and achieved local sample burnups from 45 to 56 GWd/MTU over two irradiation cycles. Rods O1, O12, and O13 had an initial enrichment of 4.657 wt% ^{235}U and achieved burnups between 22 and 30 GWd/MTU in one irradiation cycle.

The MALIBU international program was coordinated by the Belgian company Belgonucleaire. The assay measurements were performed on three spent fuel samples, selected from the same fuel rod irradiated in the Gösgen PWR operated in Switzerland, with 4.3 wt ^{235}U initial enrichment. Two fuel rod samples were selected from adjacent positions near the end of the fuel rod, and had burnups of approximately 46 and 51 GWd/MTU. The third sample was from a central position and had a burnup value close to 70 GWd/MTU. Isotopic measurements of about 50 isotopes were performed for each sample and included uranium, plutonium, neptunium, americium, curium, and many fission products important to spent fuel criticality safety. The program was designed to provide reliable data with low experimental uncertainties, and it achieved this objective by using high precision measurement techniques and performing the measurements at three independent radiochemical analysis laboratories. The data from the MALIBU program represents some of the highest quality isotopic data currently available for high burnup fuel.

ARIANE, an international program designed to improve the database of isotopic measurements for spent fuel source term and isotopic inventory validation, was coordinated by Belgonucleaire and completed in March 2001. This collaborative project involved participants from laboratories and utilities from seven countries: Belgium, Germany, Japan, Netherlands, Switzerland, the United Kingdom, and the United States. A key feature of the ARIANE program was that three cross-checking laboratories performed the measurements to reduce the experimental uncertainties and improve confidence in the measured data: Studiecentrum voor Kernenergie - Centre d'Étude de l'Énergie Nucléaire (SCK-CEN) in Belgium, Paul Scherrer Institute (PSI) in Switzerland, and Institute for Transuranium Elements (ITU) in Germany. Measurements were carried out on both UO_2 and MOX fuel between 1996 and 1999. The three UO_2 samples were selected from fuel rods irradiated in the Gösgen reactor operated in Switzerland and range in burnup from approximately 30 to 60 GWd/MTU. One of these samples was obtained from an assembly with an initial enrichment of 3.5 wt ^{235}U that was irradiated for four consecutive cycles. The other two samples, irradiated for three cycles, were taken from a rebuilt assembly with initial fuel enrichment of 4.1 wt% ^{235}U.

The REBUS International Program, coordinated by Belgonucleaire, was dedicated to the validation of computer codes for criticality calculations that take into account the reduction of reactivity of spent fuel as a result of burnup. Participants in REBUS included institutes from Belgium, France, Germany, Japan, and the United States. Completed in December 2005, REBUS involved critical measurements in a critical facility at SCK-CEN using spent fuel rod segments. One of the segments was assayed to determine the isotopic content of the fuel. The results for this sample, measured by the SCK-CEN laboratory in Belgium, were reported as part of the program. The sample was obtained from a fuel rod of an 18 X 18 PWR assembly operated in the German reactor Gemeinschaftskernkraftwerk Unit II in Neckarwestheim/Neckar. Although this reactor currently operates with a MOX core, the assembly was obtained from the reactor during a period when it operated with only UO_2 fuel. The measured sample had an initial enrichment of 3.8 wt% and a burnup of about 54 GWd/MTU.

Since the issuance of ISG-8, Rev. 2, in 2002, depletion codes which can model the non-uniformity of the spent fuel assemblies (e.g., water holes, burnable absorber rods) in predicting an accurate isotopic inventory have become available. For example, the NRC staff will examine the results of the above chemical assay programs using the two-dimensional TRITON depletion control module of the SCALE computer code in considering the extension of burnup credit beyond that recommended in ISG-8, Rev. 2.

With respect to computational techniques to determine the propagation of isotopic biases and uncertainties, those approaches which have been discussed in ISG-8, Rev. 2, will be reconsidered. Realistic conservatism in depletion modeling and design-independent isotopic biases and uncertainties are two considerations for revising ISG-8. Rigorous statistical analysis of the conservatisms in the irradiation parameters assumed for the depletion analysis, as well as the methodologies used to develop isotopic biases and uncertainties, may produce a more realistic approach to depletion analysis, while maintaining an adequate safety margin. Design-independent isotopic biases and uncertainties would allow a package designer to analyze multiple designs without going through the entire isotopic validation process in each application, provided the same depletion code and isotopic cross sections are used. These approaches will be examined by NRC staff in developing ISG-8, Rev. 3.

III.B Criticality Benchmarking

Since the release of ISG-8, Rev. 2, new domestic and international critical experiment data have become available. Currently, there are three major sources of data with respect to benchmarking the transportation package criticality safety computer codes used for Burnup Credit analyses. These sources are a) the IHECSBE; b) Commercial Reactor Critical (CRCs) state points, which consist of data measured at various critical states of operating reactors; and c) the French Haut Taux De Combustion (HTC) Critical Experiment Data(Ref.16).

For spent fuel transportation packages evaluated assuming fresh fuel, the IHECSBE provides a large set of criticality safety benchmark experiments for code benchmarking. For burnup credit criticality analyses, however, suitable criticality safety benchmark experiments are very limited. This is particularly true for package designs seeking burnup credit for both actinides and fission products. In general, low enrichment UO_2 and low Pu/(Pu+U) criticals are useful for BUC cask code benchmarking because they have higher similarities to the compositions of the spent fuel casks. Most of the Mixed Oxide (MOX) systems in the IHECSBE contain plutonium. The abundances of the plutonium are, in general, in the range of 6 to 9 wt%, which is quite different from the actinide content in typical spent fuel. In addition, these experiments do not contain any fission products. These differences make the MOX critical systems much less valuable to spent fuel transportation package designs seeking burnup credit for fission products. Consequently, both the industry and the NRC faces challenges in determining the criticality safety of burnup credit transportation packages due to a lack of adequate data for computer code benchmarking.

There are some fission product critical experiments data in IHECSBE. They include a subset of the French fission product criticals for Sm-149, the Sandia National Laboratories critical experiments for Rh-103, and the critical experiments performed in Japan containing natural Sm, Cs, Rh, and Eu.

Another possible source of data is the CRC state points. The benefits of using CRC data are: 1) these state points, except those at the beginning of cycle, contain fuel assemblies that have very high similarity to the contents of spent fuel packages, 2) the CRC state points provide an integral (i.e., isotopic depletion and criticality) benchmark for the criticality safety evaluation methodology, and 3) the data are more readily available. In addition, the soluble boron concentrations at the end of cycle state points are all close to zero. This condition relieves the influence of soluble boron, which must otherwise be dealt with when using the CRC state points for code benchmarking.

In addition to the above sources, approximately 150 experiments are also available from the French HTC program. This program, begun in the 1980s, was conducted by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) at the experimental criticality facility in Valduc, France. These experiments covered a variety of conditions, including arrays in pure water and poisoned water, pool storage configurations, consolidated pool storage configurations, transport cask configurations, and mixed arrays of HTC and UO_2 pins. Oak Ridge National Laboratory (ORNL) evaluated five of the HTC arrays in pure water. The pins in the arrays were 0.95 cm in diameter with a [98.9 % U(1.56) – 1.1 % Pu]O_2 composition. The plutonium-to-uranium ratio and the isotopic compositions of both the uranium and plutonium were designed to be similar to what would be found in a typical pressurized-water reactor (PWR) fuel assembly that initially had an enrichment of 4.5 wt% 235U and was burned to 37,500 MWd/MTU. The fuel material also includes ^{241}Am, which is present due to the decay of ^{241}Pu. The pin pitch varied between 1.3 and 2.3 cm. Since not all detail was publicly available, the ORNL evaluations were performed with approximate models.

For the ISG-8 Rev.3, the staff is currently considering to go beyond actinide credit by taking into account the additional data that have become available from isotopic assay programs, French HTC program for actinide criticals, and the available critical experiments for fission products. Although the available fission product criticals are very limited, the staff is considering approaches such as propagating the uncertainties associated with fission product cross sections in the current cross section libraries in terms of Δk_{eff} and comparing to the uncertainties based on the publicly available fission product criticals.

IV. SUMMARY

ISG-8, Rev. 2, represented as much burnup credit flexibility as could be expected at the time (UO_2 fuel irradiated in PWRs only, with no credit for fission products) based on the extent and range of the available data. Given the data for isotopic depletion and criticality code validation that has become available in the past several years, NRC staff has been evaluating the possibility of extending the technical basis for burnup credit in spent fuel transportation. This revised technical basis may recommend some level of credit for the minor actinides and fission products that were excluded from the recommendations of ISG-8, Rev. 2, based on a lack of data.

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Table 1 Summary of selected PWR spent fuels radiochemical assay data for ISG 8, Rev. 2

Reactor	Lattice type	Enrichment (wt%)	Burnup (GWd/t)	Absorbers	No. of samples
Trino Vercellese	WE 15 X 15	3.13	11.5 – 24.5	CR[a]	13
		3.897	12.0		1
Turkey Point	WE 15 X15	2.556	30.5 – 31.5		5
Obrigheim	CE 14 X 14	3.13	25.9 – 29.5		5
H.B. Robinson-2	WE 15 X15	2.561	16.0 – 31.7	BPR[b]	4
Yankee Rowe	WE 17 X 18	3.4	16.0 – 36.0	CR	8
Calvert Cliffs	CE 14 X 14	3.038	27.4 – 44.3		3
		2.72	18.7 – 33.2		3
		2.453	31.4 – 46.5	BPR	3
Takahama-3	WE 17 X 17	4.11	14.3 – 47.3	BPR	10
Range		.56 – 4.11	11.5 – 47.3		Total 56

CR[a] = Assemblies exposed to control rods

BPR[b] = Assemblies with burnable poison