Advisory and Assistance Services Task Order #2

Technical Data to Justify Full Burnup Credit in Criticality Safety Licensing Analyses

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Executive Summary

Enercon Services, Inc. (ENERCON) was requested under Task Order #2 to identify scientific and technical data needed to benchmark and justify Full Burnup Credit, which adds 16 fission products and 4 minor actinides[1] to Actinide-Only burnup credit. The historical perspective for Full Burnup Credit is discussed, and interviews of organizations participating in burnup credit activities are summarized as a basis for identifying additional data needs and making recommendation.

Input from burnup credit participants representing two segments of the commercial nuclear industry is provided. First, the Electric Power Research Institute (EPRI) has been very active in the development of Full Burnup Credit, representing the interests of nuclear utilities in achieving capacity gains for storage and transport casks. EPRI and its utility customers are interested in a swift resolution of the validation issues that are delaying the implementation of Full Burnup Credit [EPRI 2010b]. Second, used nuclear fuel storage and transportation Cask Vendors favor improving burnup credit beyond Actinide-Only burnup credit, although their discussion of specific burnup credit achievements and data needs was limited citing business sensitive and technical proprietary concerns. While Cask Vendor proprietary items are not specifically identified in this report, the needs of all nuclear industry participants are reflected in the conclusions and recommendations of this report.

In addition, Oak Ridge National Laboratory (ORNL) and Sandia National Laboratory (SNL) were interviewed for their input into additional data needs to achieve Full Burnup Credit. ORNL was very open to discussions of Full Burnup Credit, with several telecoms and a visit by ENERCON to ORNL. For many years, ORNL has provided extensive support to the NRC regarding burnup credit in all of its forms. Discussions with ORNL focused on potential resolutions to the validation issues for the use of fission products. SNL was helpful in ENERCON's understanding of the difficult issues related to obtaining and analyzing additional cross section test data to support Full Burnup Credit.

A PIRT (Phenomena Identification and Ranking Table) analysis was performed by ENERCON to evaluate the costs and benefits of acquiring different types of nuclear data in support of Full Burnup Credit. A PIRT exercise is a formal expert elicitation process with the final output being the ranking tables. The PIRT analysis (Table 7-4: Results of PIRT Evaluation) showed that the acquisition of additional Actinide-Only experimental data, although beneficial, was associated with high cost and is not necessarily needed. The conclusion was that the existing Radiochemical Assay (RCA) data plus the French Haut Taux de Combustion (HTC)[2] and handbook Laboratory Critical Experiment (LCE) data provide adequate benchmark validation for Actinide-Only Burnup Credit.

The PIRT analysis indicated that the costs and schedule to obtain sufficient additional experimental data to support the addition of 16 fission products to Actinide-Only Burnup Credit

[1] For the purposes of this report, the reactivity benefit of 16 fission products and 4 major actinides will collectively be referred to as fission product burnup credit.
[2] The French HTC experiments are limited rights data and require a Non-Disclosure Agreement in place that specifies the terms of use.

to produce Full Burnup Credit are quite substantial. ENERCON estimates the cost to be $50M to $100M with a schedule of five or more years.

The PIRT analysis highlights another option for fission product burnup credit, which is the application of computer-based uncertainty analyses (S/U – Sensitivity/Uncertainty methodologies), confirmed by the limited experimental data that is already available. S/U analyses essentially transform cross section uncertainty information contained in the cross section libraries into a reactivity bias and uncertainty. Recent work by ORNL and EPRI has shown that a methodology to support Full Burnup Credit is possible using a combination of traditional RCA and LCE validation plus S/U validation for fission product isotopics and cross sections. Further, the most recent cross section data (ENDF/B-VII) can be incorporated into the burnup credit codes at a reasonable cost compared to the acquisition of equivalent experimental data. ENERCON concludes that even with the costs of code data library updating, the use of S/U analysis methodologies could be accomplished on a shorter schedule and a lower cost than the gathering of sufficient experimental data. ENERCON estimates of the costs of an updated S/U computer code and data suite are $5M to $10M with a schedule of two to three years.

Recent ORNL analyses using the S/U analysis method show that the bias and uncertainty values for fission product cross sections are smaller than previously expected. This result is confirmed by a similar EPRI approach using different data and computer codes.

ENERCON also found that some issues regarding the implementation of burnup credit appear to have been successfully resolved especially the axial burnup profile issue and the depletion parameter issue. These issues were resolved through data gathering activities at the Yucca Mountain Project and ORNL.

As a result, ENERCON concludes there are two basic paths-forward for obtaining fission product data: a longer-term data acquisition program to support traditional validation of Full Burnup Credit with LCE and RCA data, and a shorter-term program using S/U analysis methods to provide validation of Full Burnup Credit with computer analyses and existing confirmatory data. An advantage of the S/U analysis method is that it allows the NRC to evaluate Full Burnup Credit in the near term, while retaining the possibility of improving the approved credit in the long-term.

ENERCON recommends that the S/U methodology be used to add fission products to the Actinide-Only Burnup Credit to achieve Full Burnup Credit in the next 2-3 years. We further recommend that existing fission product data, plus the results of any programs that will yield results in the 2-3 years, be used to validate the S/U calculations. ENERCON believes the nuclear industry would prefer to improve upon the Actinide-Only Burnup Credit as much as possible in the next 2-3 years and that further delay to pursue gathering a full set of experimental data on all 16 fission products and 4 minor actinides is not desirable.

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Acronyms

ANL	Argonne National Laboratory
BWR	Boiling Water Reactor
CRC	Commercial Reactor Critical
DOE	Department of Energy
ENDF/B	A nuclear data file format maintained by a group of National Laboratories
ENDF/B-VII	The latest version of ENDF/B
EPRI	Electric Power Research Institute
FP	15 Principal Fission Products plus Cesium-133
GRESS	A general purpose code system adapted to apply the SAS2H code sequence to evaluate the reactivity effect of isotopic uncertainties engendered by cross section uncertainties
GWd/MTU	GigaWatt-day/Metric Ton Uranium
HTC	Haut Taux de Combustion
LCE	Laboratory Critical Experiment
LWR	Light Water Reactor
MCNP	Monte Carlo N-Particle computer code
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ORIGEN-S	A computer code for calculating isotopic compositions of spent fuel
ORNL	Oak Ridge National Laboratory
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
RCA	Radiochemistry Assay
SAS2H	A computer code sequence for performing one dimensional depletion calculations to generate isotopic compositions of spent fuel
SCALE	Standardized Computer Analyses for Licensing Evaluations
S/U	Sensitivity/Uncertainty, a methodology to transform uncertainty information contained in ENDF/B into bounding reactivity uncertainties
TSUNAMI	One of the computer code suites used to perform S/U calculations
TRITON	A modern replacement to the SAS2H code capable of performing two dimensional depletion calculations to generate isotopic compositions of spent fuel
YMP	Yucca Mountain Project

1 Introduction and Scope

ENERCON was requested under Task Order # 2 – "Technical Data to Justify Full Burnup Credit in Criticality Safety Licensing Analysis" to identify the scientific and technical data needed to benchmark and justify burnup credit, including credit for the principal actinides and fission products and minor actinides products[3], in criticality safety licensing analyses involving commercial used nuclear fuel, including new generation fuel. Burnup credit allows the design of high capacity transport casks, storage casks and disposal packages for used nuclear fuel.

The technical data needs include:

- RCAs to provide measured isotopic compositions of used nuclear fuel to support depletion code/data validation
- LCEs, including LCEs with fission products, to support criticality code/data validation
- Fission product and minor actinide cross-section measurements for selected nuclides, as needed, to improve confidence in reactivity predictions of the fission products
- A database of reactor operational history information to support justification of depletion parameters and assumptions used in the licensing safety analysis.

The scope of work includes review of existing burnup credit work and soliciting industry perspective on data needs that would improve efficiency of packaging used nuclear fuel, ease licensing, and reduce costs related to obtaining burnup credit to meet criticality requirements for transportation, storage, and disposal licensing.

This report provides a review of the current status of burnup credit in the nuclear industry and the national laboratories and the additional data needed to obtain Full Burnup Credit; a description of a PIRT evaluation applied to this subject and the results of the evaluation; an estimate of cost and schedule for the processes identified for obtaining data for Full Burnup Credit; a discussion of conclusions; and ENERCON's recommendation for further action.

2 Background

The NRC has approved spent fuel storage/transport casks with burnup credit using Actinide-Only (Nuclear Assurance Corporation, pending), and Full Burnup Credit (limited credit, HOLTEC Corporation, approved). The HOLTEC approval included fission products, but was based upon proprietary analysis methodologies and insufficient data was available for the NRC to grant full credit for the fission product inventories. The approval of Actinide-Only burnup credit is straightforward due to the acquisition by DOE of the French HTC Critical Experiment Data.

[3] For the purposes of this report, the reactivity benefit of 16 fission products and 4 major actinides will collectively be referred to as fission product burnup credit.

The HTC data has been evaluated by ORNL [Mueller 2008] and can be combined with existing published laboratory critical data to validate all uranium, plutonium, and MOX configurations of spent nuclear fuel.

Published RCA data for spent fuel pellets is adequate to validate the uranium and plutonium isotopic contents of spent fuel. Thus sufficient data has been developed for Actinide-Only burnup credit.

Actinide-Only burnup credit refers to calculations employing only uranium (^{234}U, ^{235}U, ^{236}U, and ^{238}U) and plutonium (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) isotopes. "Full" Burnup Credit refers to a combination of the uranium and plutonium isotopes evaluated in Actinide-Only burnup credit, plus a number of fission products and minor actinides. In this report, "Fission Product" or "FP" burnup credit is used to refer to 16 fission product isotopes (^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs , ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{151}Eu, ^{153}Eu,^{155}Gd) and 4 minor actinide isotopes (^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am). Different numbers of fission product isotopes have been used by different researchers in the burnup credit field, but the consensus is that the most important six are ^{149}Sm, ^{103}Rh, ^{143}Nd, ^{151}Sm, ^{155}Gd, and ^{133}Cs. ^{241}Am is the most significant minor actinide, contributing a reactivity worth (as a neutron absorber, this is beneficial to burnup credit) that is ranked third or fourth among the fission product isotopes. ^{237}Np is also significant, although not as strong in terms of reactivity worth as the six fission products and ^{241}Am.

The French HTC critical experiments used uranium and MOX fuel pins. Experiments were also performed to confirm the reactivity effect of the six most important fission products, but acquisition of this data has been difficult. A limited number of LCEs have been performed in Japan which are applicable to fission product burnup credit calculations.

LCE data is thus limited for fission products. Similar limitations are found for RCA data that include fission products. The most complete RCA dataset is for TMI-1 spent fuel, but this data apparently contains a systematic bias relative to all other RCA data and consequently it has not been accepted for use to substantiate Full Burnup Credit. The French nuclear program has provided some very limited fission product RCA data.

An alternative to the LCE and RCA data is the CRC (Commercial Reactor Critical) data, which consists of forty-one reactor restart and four initial startup physics tests, which are integral experiments that verify the combined effect of isotopic content and cross section data. The CRC data supports Full Burnup Credit but is not a preferred primary data source.

3 Methodology

The methodology to identify the technical needs is as follows:

- Review historical documents, summarize status and current limitations in the data
- Review the primary data needs with the current US stakeholders
- Review the primary data needs with foreign stakeholders

- Develop a report identifying data needs, rationale, approach to gathering data (including organization best suited to gather data), estimated cost and schedule required to gather data. The data needs are prioritized using a PIRT process.

4 Data Types

There are several basic forms of data needed to perform burnup credit calculations:

- Nuclear data libraries, principally ENDF/B. Burnup Credit calculations have been performed with ENDF-B/VI and earlier libraries. The recent release of ENDF/B-VII offers improved data for burnup credit calculations.
- Validation data to develop the bias and uncertainty of an application. Full Burnup Credit requires validation data for the isotopic contents of spent fuel and validation data for the nuclear cross sections used in the calculation of k_{eff}.
- Input data, to define the irradiation history of the spent fuel itself, so that the isotopic content in the fuel is accurate. The axial profile of the burnup is an issue that has required considerable effort to resolve, but acceptable solutions are now available. The fuel depletion parameters that define the fuel and moderator temperatures, specific power and time history, dissolved boron content, and fuel assembly geometry data have also been studied, and guidance is available for the various parameters.

The relationship of these different data types and the computer codes used for burnup credit calculations are illustrated below. The input data defines the irradiation history and environment that the fuel experiences while it is "burning". The axial burnup profile defines the burnup that will be achieved at various positions down the length of the fuel assembly, with a peak in the central length and lower burnups at the top and bottom ends. The reactivity of spent fuel is increased when the burnup profile has lower-than-average burnups at the ends, so conservative representations of the axial profile are used.

4.1 Axial Burnup Profiles

An example of a burnup profile for a PWR spent fuel assembly is shown in Figure 4-1. The original data is defined for 18 axial node positions, which is reduced to 7 to provide a more compact conservative representation [Scaglione 2003]. The top and bottom three nodes are defined by using a statistical tolerance limit approach at a 95 percent confidence level for 95 percent of the population being above the smallest observed burnup value, and the central node is adjusted to preserve the total burnup for the fuel assembly. The result is a statistically bounding profile for PWR spent fuel with initial enrichments up to 5.0 wt%. An ORNL report also provides guidance for axial profiles for PWR spent fuel with initial enrichments up to 4.0 wt% [Wagner 2003]. A similar axial profile is provided for BWR spent fuel [Huffer 2004].

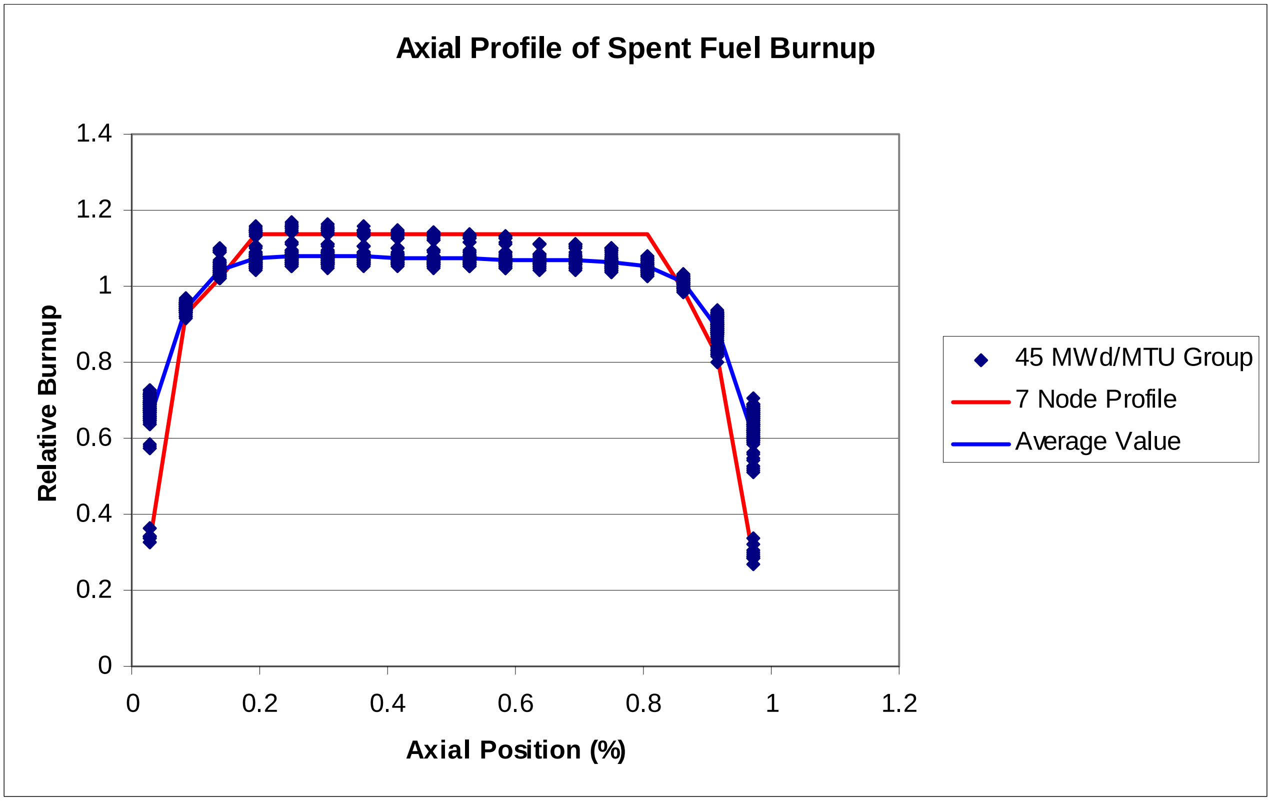

Figure 4-1: Axial Profile of Spent Fuel Burnup

4.2 Fuel Depletion Parameters

The isotopic contents determined by the depletion code (e.g. TRITON or SAS2H) are passed to the reactivity calculation code (e.g. MCNP or KENO), which uses a description of the burned fuel plus the support structure (fuel basket or rack) and any strong neutron absorbers inserted into the structure. Some examples of PWR and BWR depletion parameters are shown in Table 4-1. The ranges and trends of the various parameters are evaluated in ORNL [DeHart 1996] and the YMP [Scaglione 2001]. In actual use by an applicant, the bounding parameters are selected based upon the particular spent fuel contents of a cask or rack.

Table 4-1: Comparison of Bounding and Nominal Depletion Parameters

	PWR	PWR	BWR	BWR	
Parameter	**Bounding**	**Nominal**	**Bounding**	**Nominal**	
T_{fuel}	1200	861.3	1200	1000	ºK
$T_{moderator}$	602.6	579.8	560.7	560.7	ºK
moderator density	0.6516	0.7556	0.30	0.43	g/cm^3
Power	30.00	43.029	22.38	35.68 avg.	MW/MTU
Dissolved Boron	950	Varies	None		ppmB
UO_2 density	10.741	10.121	10.741	10.121	g/cm^3
Burnable Poisons	Yes	Yes	Yes	Yes	
Control Rods	No*	No*	Yes**	Yes**	

* Burnable Absorbers in All Cycles to Bound Effects of Control Rods
** Control Blades Inserted in Final Cycle to Maximize Pu Production

The Evaluated Neutron Data File, ENDF/B-VI or earlier versions, provides the neutron data necessary for the reactivity code to calculate k_{eff}. The most recent version of ENDF/B, version VII, has improvements important to burnup credit calculations and is desirable.

4.3 Burnup Credit Data Flow

The overall burnup credit data flow is illustrated in Figure 4-2.

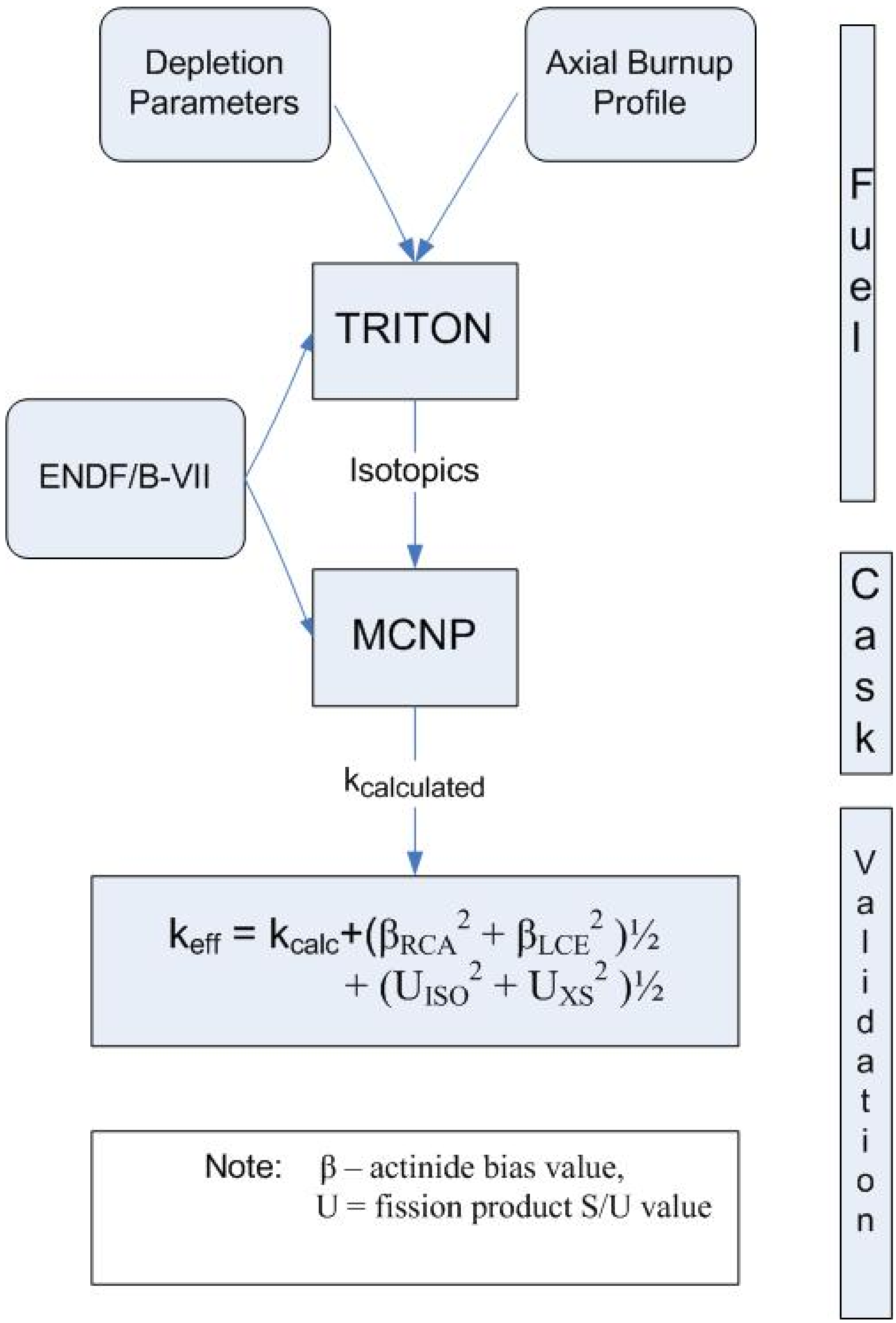

Figure 4-2: Burnup Credit Data Flow

Once k_{eff} is calculated, it is adjusted to account for the presence of bias and uncertainty in the various components of the calculations. The Actinide-Only terms, represented as β_{RCA} and β_{LCE} in the Figure, are the isotopic bias and uncertainty determined from evaluations of RCAs and the cross section bias and uncertainty determined from evaluations of LCEs. The methods and data used for the Actinide-Only bias and uncertainty determinations are well-understood, although improvements in the definition of RCAs for use by the two-dimensional TRITON computer code system are desirable. The detailed geometry of fuel rods in the assembly can be modeled exactly by TRITON but not SAS2H, so approximations were made that should be updated.

The validation of Actinide-Only burnup credit using RCAs and LCEs is illustrated in a schematic diagram in Figure 4-3. The schematic diagram shows that RCAs are used to provide a bias and uncertainty term that are used to validate the isotopic concentrations calculated by TRITON or SAS2H. The isotopic bias and uncertainty are calculated as a Δk_{eff} term that may be used in the determination of the Upper Safety Limit (USL), which is the maximum value of the calculated k_{eff} that satisfies all of the limits for the calculation.

Conceptually, Full Burnup Credit may be considered as the combination of Actinide-Only burnup credit and Fission Product burnup credit, where Fission Product burnup credit is the reactivity worth of the additional 16 fission products and 4 minor actinides to the isotopic mix of the major actinides of U and Pu.

The Fission Product bias and uncertainty terms are represented as U_{ISO} and U_{XS} in Figure 4-2, indicating their evaluation through S/U methodologies. The traditional approach would be to include the β_{RCA} (FP) and β_{LCE} (FP) for the fission products, if sufficient RCA data and/or LCE data were available. Lacking sufficient RCA and LCE data at the current time, the S/U methodologies for fission product isotopic and cross section data could be employed as shown in Figure 4-4, with the intent of applying a traditional RCA or LCE approach at a later date when sufficient data becomes available. Note that there are three physical parts of the isotopic S/U validation to represent the effects of the uncertainty of fission yield, radioactive decay, and neutron capture. The neutron capture term has been evaluated using the GRESS computer software and the fission yield and radioactive decay terms can be evaluated by Direct Perturbation calculations [Wells 2006 and Connell 2002].

Note also that NRC ISG-8 Rev. 2 [NRC 2002] suggests that an applicant include a calculation of the “uncredited margin” of fission products when the applicant submits an Actinide-Only Burnup Credit analysis. In this case, the S/U methodology can be used to provide the fission product reactivity worth, clearly defining the “extra” margin present in the Actinide-Only approach.

The S/U methodologies for cross sections and isotopics are well-understood, but could require updating to use the ENDF/B-VII data since many analyses were developed with earlier datasets.

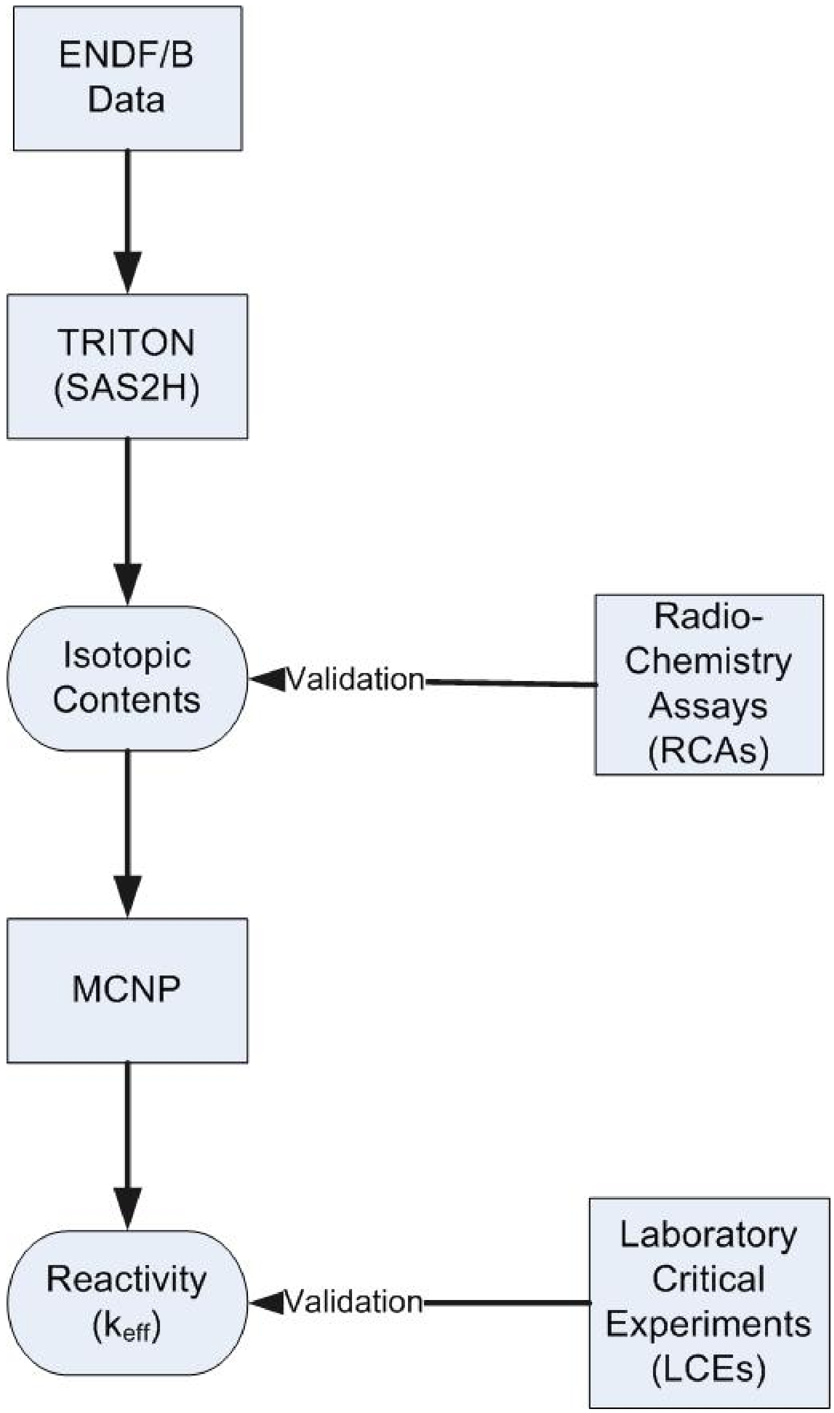

Figure 4-3: Application of RCA and LCE Data to Actinide-Only Burnup Credit

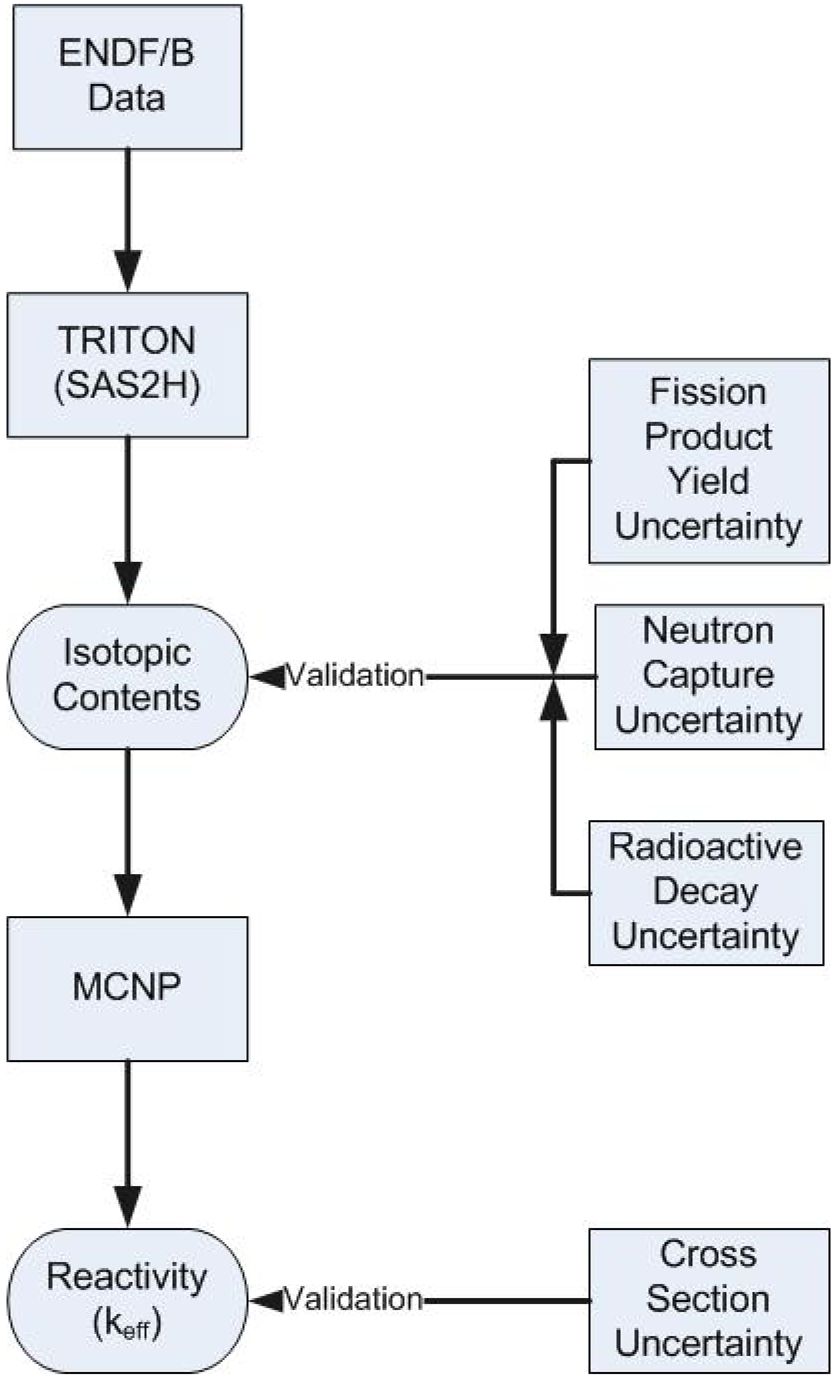

Figure 4-4: Application of S/U Data to Fission Product Portion of Burnup Credit

5 Review of Burnup Credit

There have been a number of recent publications that address various issues for burnup credit. Some of these relate to the status and potential path-forward for Full Burnup Credit and are briefly described below. A helpful link for burnup credit references is: http://www.ornl.gov/sci/radiation_transport_criticality/BUCPublications.htm

Following are summaries of the discussions that the preparers have had with various organizations related to the review of burnup credit.

5.1 Cask Vendors

Cask Vendors have been reticent regarding their burnup credit efforts due to the proprietary nature of their licensing submittals. However, they have expressed continuing interest in burnup credit, as manifested in licensing applications that have been submitted. This indicates that they perceive a value to Full Burnup Credit. ENERCON has evaluated the needs of Cask Vendors and has factored them into the recommendations of this document.

5.2 Industry Organizations

The Nuclear Energy Institute (NEI) and EPRI, both represent the interests of the nuclear utility industry, have pursued burnup credit for many years. EPRI participated in efforts to obtain the HTC and fission product critical experiment data from France, and continues its efforts to acquire RCA data for fission products. EPRI has evaluated the incremental risk of a criticality event facilitated by the use of burnup credit relative to the risks of storage and transport of spent fuel, and has found the additional risk to be minimal [EPRI, 2010b]. EPRI has also developed an approach for Full Burnup Credit [EPRI, 2010a] using methodologies similar to the S/U methods developed by ORNL. Thus, EPRI recommends the implementation of Full Burnup Credit and would like to see it available to nuclear utilities in the near term.

5.3 National Laboratories

5.3.1 SNL

SNL has been active in the development of Full Burnup Credit dating back to the early 1980s. SNL has performed evaluations of the risks of burnup credit based relative to the risk of all spent fuel shipping activities, and evaluated the additional risk of burnup credit as minimal.

SNL has also been active in the measurement of fission product cross section bias through a series of experiments using enriched reactor fuel that was used in extensive criticality experimentation at Hanford [Harms 2009]. The experimental procedure is to place foils of isotopically pure fission product material between the fuel pellets of a number of fuel rods. These experiments are of good quality but the tasks of reconstructing the rods with new foils for each isotopic measurement have been expensive and progress has been slow.

5.3.2 ORNL

ORNL has been the most active of the national laboratories for investigations of various forms of burnup credit, and has provided the evaluations of the HTC actinide-only critical experiment data acquired from France [Mueller 2008].

ORNL has performed extensive analyses of the sensitivity of spent fuel reactivity due to changes in the depletion parameters used to describe the environment of fuel assemblies as they are burned in the reactor. These are summarized as follows:

Depletion Parameters – ORNL has performed extensive studies of the effects of variations of the depletion parameters that define how a fuel assembly was burned. The studies focused on a single parameter at a time, and did not consider inter-relationships between depletion parameters. Thus, the procedure to date has been to select "bounding" values for each parameter, resulting in conservative isotopic contents.

BPRA Modeling –The use of Burnable Poison Rod Absorbers (BPRAs) in a fuel assembly results in a hardening of the neutron spectrum and an increase in the production and burning of plutonium isotopes, with the result that a discharged fuel assembly has a higher reactivity. ORNL has performed extensive studies of the BPRA effects and selected a conservative set of operating parameters to insure a conservative result.

TSUNAMI Analyses of Fission Product Laboratory Critical Experiments – ORNL has performed analyses of the cross section uncertainties of the 16 fission products for Full Burnup Credit [Mueller 2009]. The S/U methodologies employed for these analyses showed that the cross section uncertainties have a smaller reactivity effect than previously expected. The ORNL researchers conclude: "From this analysis, it is unlikely that the bias associated with the 16 FPs evaluated would be as large as 0.13% $\Delta k/k$, which is less than 2% of the FP worth from the model." This approach has the potential for resolving the difficulties caused by the scarcity of fission product LCEs.

RCA Data Analyses – ORNL has produced many reports over the years that dealt with the validation of the SAS2H module of the SCALE system. SAS2H is a one-dimensional analysis of the burning of fuel with surrounding moderator in a cylindrical geometry. In order to represent two dimensional features of fuel assemblies, such as guide tubes and BPRAs, SAS2H incorporated a two-pass processing approach that created a one-dimensional model of the fuel rod/moderator cells and a one-dimensional model of the BPRA/moderator cells, and then weighted them based upon the relative number of BPRAs and fuel rods and the relative strengths of the thermal, resonance, and fast fluxes.

TRITON – ORNL has developed a two-dimensional replacement to SAS2H. The original SAS2H validation RCA data must be re-evaluated using TRITON. The original SAS2H models were created by simulating the two-dimensional features of a fuel assembly with weighted one-dimensional data, but this process is one-way: one cannot retrieve the original two-dimensional data from the SAS2H model. The isotopic contents of some fuel pellets are quite sensitive to the local moderator and neutron absorber materials, so accurate knowledge of the 2D geometry and materials must be re-established from the original source reports. The original data sources for the isotopic validation datasets must be re-evaluated for use in TRITON, and updated as necessary.

CRC Analyses – ORNL performed an in-depth study of the CRC database which showed that the startup physics tests, the CRC experiments, are relevant to the validation of criticality calculations for spent fuel casks and waste packages. The spectra in the CRCs and the containers are quite similar, and the interactions of each isotope proceed at similar relative rates. This similarity is an important requirement for validation efforts.

Other Data Analyses – ORNL has also had access to other validation projects (MALIBU, REBUS, and French fission product critical experiments). Most of this data is proprietary and a methodology to incorporate all of these data types has not been established.

5.4 Yucca Mountain Project

The Yucca Mountain Project (YMP) was very active in the development of Full Burnup Credit and performed data analyses similar to those performed at ORNL in many cases.

5.4.1 CRC Database

One unique data source obtained by the YMP is the CRC database. This database contains detailed information of each fuel assembly that was burned in a reactor during an operational cycle, so that the reactivity of the reactor at the time of restart (or startup) could be accurately calculated. There were 40 restart experiments and 4 fresh startup experiments, for a total of 44 datasets. The positions of the control rods and dissolved boron concentration when the reactor is restarted are measured and compared to calculated positions each time a reactor is started to insure that the core is properly loaded.

The CRC database is useful because it is a test of actual spent fuel, mixed with new fuel as the core is shuffled. This use of actual spent fuel is unique, as laboratory critical experiments with spent fuel are quite rare. The macroscopic cross section for neutron capture and scattering (i.e. by the hydrogen in water) is validated by CRCs, but the individual breakdown of isotopic contents and cross sections for each isotope is not possible. Thus, the CRCs are called "integral" experiments.

The CRC dataset provides several unique and valuable perspectives on Full Burnup Credit. The CRCs show that the ENDF/B-VI and ENDF/B-V datasets and the MCNP computer code under predict the reactivity of the critical reactor by a Δk of 0.0062 with an uncertainty of 0.0061, which produces an overall Bias factor of -0.0068 [Wells 2003].

5.4.2 Axial Profiles ($T_{95/95}$)

The YMP developed a statistically based model for spent fuel assembly axial burnup profiles. The concept is to avoid the issues of "worst case" axial profiles by providing an axial profile such that, at the top three and bottom three axial nodes, the profile bounds 95 percent of the data points 95 percent of the time. The axial database used to develop the profiles contains enough data to allow the profiles to be broken down by fuel initial enrichment and also into burnup groups. The data extends to 5.0 wt% so that it exceeds the 4.0 wt% limit used in some previous ORNL evaluations of axial burnup profiles.

5.4.3 Sensitivity of Reactivity to Cross Section Uncertainties

The sensitivities of k_{eff} to uncertainties in the cross sections of fission products and minor actinides were determined by Direct Perturbation calculations performed with MCNP using LANL ENDF review data [Bowen 2004]. The LANL data provided a review of the thermal cross section uncertainties for all of the burnup credit nuclides, including the major actinides, fission products, and minor actinides. A YMP study [Wells 2005] demonstrated that the thermal cross section uncertainties dominate the overall uncertainties by showing that the interaction rate in each isotope is much stronger at thermal neutron energies. This result is expected since LWR fuel is well moderated. For those isotopes which do not have ENDF thermal cross section uncertainty values, an alternative approach was used where the various published values of the thermal cross section over all ENDF versions through ENDF-B/VI variants are treated as if they were provided by a group of experts. The use of expert data was acceptable to the YMP QA Plan, and the reported cross sections were evaluated to determine the range of values, which was then expressed as an uncertainty. The Direct Perturbation calculations were performed by running separate MCNP calculations for each fission product and minor actinide isotope with the unperturbed thermal cross section and then with the cross section adjusted by the percentage value of the uncertainty. The actual adjustment was performed by adjusting the isotopic content instead of the microscopic cross section since this yields a properly adjusted macroscopic cross section without requiring modifications to the ENDF libraries. The differences between the unperturbed and perturbed k_{eff} are calculated and typically expressed as percentages of the worth of each fission product or minor actinide. The sum of all fission product and minor actinide isotope uncertainties represents the "loss" of worth of the criticality control that can be provided by these absorbers.

5.4.4 Sensitivity of Reactivity to Isotopics

The GRESS computer code system was developed at ORNL to allow the sensitivity (S) of isotopic contents to changes in cross sections. The isotopic contents are determined by the fission yield curve, the small effect of radioactive decay, and the effect of transmutation of isotopes produced by fission by neutron capture. Uncertainties in neutron capture cross sections affect the predicted isotopic contents of spent fuel as absorbers are burnt out via transmutation faster because of the uncertainty while fissile materials are less affected because of their uncertainties.

The GRESS computer code (ORNL/TM-9771) calculations allow the uncertainty in cross sections to be related to an uncertainty in isotopic contents. The sensitivity S computed by GRESS may be expressed as $\Delta I/\Delta U$ for each isotope, where ΔI is a change in the inventory of an isotope and ΔU is the cross section uncertainty. If the effect of the cross section uncertainty ΔU is expressed as a fraction of the worth of any isotope, then the effect of the cross section uncertainty upon the isotopic contents results in a reactivity worth Δk_I (transmutation) of S* ΔU.

The reactivity effect of the uncertainty in the fission yield of isotopes, Δk_I(fission yield), may be obtained from a Direct Perturbation calculation (for each isotope) to determine the S/U parameter, which is then multiplied by the uncertainty of the isotope to produce the Δk_I value for the isotope. The uncertainties for the isotopes are obtained from [IAEA 2000].

The reactivity effect of the uncertainty in the radioactive decay of isotopes during the irradiation history is quite small [Connell 2002].

6 Discussions and Reviews with Industry and National Laboratories

6.1 Cask Vendors

ENERCON met with or had telephone conference calls with three Cask Vendors, i.e Nuclear Assurance Corporation (NAC), HOLTEC, and Transnuclear (TN). Each of the Cask Vendors were limited in their response to our inquiry about additional data needs to support Full Burnup Credit citing business sensitive and technical proprietary concerns.

The proprietary nature of Cask Vendor's burnup credit methodologies limits an open discussion of the details. In general, the Cask Vendors favor Full Burnup Credit over Actinide-Only Burnup Credit because of the improved criticality control capabilities of Full Burnup Credit. The resolution of the validation issues hindering Full Burnup Credit would allow a more robust loading curve and the ability to store and/or transport more fuel assemblies. The most pressing concerns are difficulties regarding the validation of Full Burnup Credit calculations. The desired timeframe for resolution of the validation issues is immediate due to commercial pressures to store and transport as much fuel as possible within each cask.

6.2 Industry Organizations

6.2.1 EPRI

Summary of Conference Call with EPRI on 1/4/2011

EPRI is interested in burnup credit because the nuclear utilities are currently loading spent fuel in casks which have lower capacities than necessary due to the inability to apply Full Burnup Credit. EPRI indicated that it is not certain that their efforts to insure that spent fuel could be transported at high burnups [EPRI, 2010b] have succeeded in convincing the NRC, because an applicant has not tried to use the EPRI work in the licensing efforts for a high burnup cask. EPRI continues to be interested in the issues of burnup credit, misloaded fuel and selection of assigned burnup for fuel loading using reactor record burnups. EPRI has developed a burnup credit approach based upon the application of S/U methods [EPRI, 2010a]. EPRI expressed an interest in cross section measurements for fission products and minor actinides, but had concerns regarding the potential for improvement of RCA data for fission products and minor actinides. The current status of the YMP has reduced utility interest in the Transport, Aging, and Disposal Canister, which benefits from burnup credit.

6.3 National Laboratories

6.3.1 SNL

Summary of Conference Call with SNL on 1/11/2011.

SNL has participated in burnup credit development efforts since the 1980s and led the earliest efforts for the YMP. Current efforts include the development of LCEs for fission products, using a reactor core configuration. The initial testing was for the ^{103}Rh fission product using thin foils inserted into select fuel rods. This methodology distributes the fission product material axially through a fuel rod and simulates the actual spent fuel rod conditions. The use of foils allows the fuel rod to be reconstituted with a variety of fission product foils, so that all of the 16 fission product isotopes could be tested. This methodology appears to produce accurate results. The fuel rods used for these experiments are un-irradiated, so that handling during the process of adding the fission product foils does not result in personnel dose constraints. This LCE test facility is available for further testing.

6.3.2 ORNL

Summary of Conference Call with ORNL on 10/7/2010

The following items were discussed:

LCE Validation Issues:

- Fission product computations predicted by nuclear data uncertainties were in line with measurements, but conservative.
- Measurements of the reactivity effects of fission products by critical experiments were expensive and not necessarily cost effective

RCA Issues:

- Actinide isotopic computations were in good shape (relatively well characterized).
- For fission products, the bias appears to be small, but the uncertainty is high.
- More fission product data is needed, but it is expensive and measurement uncertainties are an issue.
- Generally for older RCA data and even for recent data, the traceability (pedigree) of the data has been somewhat problematic.
- There is a need to understand data uncertainty and propagation.

Operational Database of Reactor Data:

- There is a need for database for depletion parameters including moderator density/temperature, soluble boron and axial burnup.

Summary of Meeting on Full Burnup Credit at ORNL on 10/15/2010

ENERCON reviewed the objective of the DOE task to identify the technical data needs for Full Burnup Credit including fission products. ORNL reviewed the analytical work that ORNL has performed on burnup credit validation, particularly recent RCA comparisons to calculations with TRITON and sensitivity/uncertainty calculations with TSUNAMI.

ORNL discussed the need to obtain more RCA data since only 10 samples out of 120 RCA data sets included a complete set of actinide and fission products at the same burnup/cooling time. Discussion continued on the difficulty and expense of performing RCAs with sufficient confidence on the reported measurements. This requires multiple laboratories performing repeat measurements of small samples. Also discussed was the difficulty of determining the exact operating conditions, i.e. moderator temperature, moderator density, fuel temperature and the neutronic influence of burnable poisons, at the sample locations.

ORNL discussed the need to identify uncertainties in reactor operating conditions for both RCAs and CRCs both of which are used to validate burnup credit methodologies. ORNL also discussed the need to quantify the similarity between CRCs and the cask environment taking credit for burnup.

Summary of Meeting on Full Burnup Credit at ORNL on 1/13/2011

ORNL mentioned that ORNL was in the process of setting up studies that could support the technical basis for approval of burnup credit including fission products over the next few years (2 to 3). The NRC will be issuing at least two NUREGS and Rev 3 of ISG-8 on the acceptable approaches to burnup credit with fission products. ORNL mentioned that the need for regulatory consistency in burnup credit evaluations for spent fuel racks, storage and transport casks was driving NRC to resolution on burnup credit including fission products. The list of 16 fission products currently under consideration was reviewed and found acceptable. Project funding for this effort was discussed and considered weak and somewhat questionable over the near term. DOE participation was considered necessary to complete the effort in the coming years.

ORNL discussed the current status of RCA and international efforts to quantify the uncertainties in the RCA measurements and the associated reactor operational data. ORNL mentioned the need for better quality data and complete actinide/fission product data for each sample. It was concluded that more data was needed, but it would take a long time and be very expensive.

ORNL and ENERCON discussed the sensitivity/uncertainty approach to quantifying the bias and uncertainties from fission products which are not adequately measured by RCA data. This approach provides the most direct and cost effective way to address burnup credit with fission products. ORNL mentioned the need to develop a new generation of software to address calculational issues and the need for refining co-variance data.

ORNL and ENERCON discussed the French fission product LCEs available and their associated cost. ORNL had recommended that the data for experiments be procured, but various attempts and funding by EPRI and DOE had failed for various reasons. They concluded that this issue should be reconsidered if possible.

ORNL mentioned the need to develop BWR burnup credit particularly for long term disposal. There was a general consensus that, while the need in storage and transport casks was not great, the need in spent fuel racks and for disposal was significant and should be developed and funded.

There was some discussion on the need for guidance on limiting operational characteristics (PWR or BWR) for depletion analysis of burnup credit isotopic. This was general consensus that guidance was needed and that there was also a need to update axial burnup profile databases because of evolving fuel designs.

The costs and level of effort for various activities were discussed in general terms. A cohesive programmatic study of the costs and possible schedules has not been made by ORNL; rather, the values discussed represent qualitative estimates.

6.4 Data Needs

Data are needed in the validation of the isotopic contents and cross sections of the 16 fission products and 4 minor actinides.

6.4.1 RCA Data

Most of the RCA datasets do not contain fission products. A few contain americium. The TMI-1 dataset includes the minor actinides and many of the important fission products, but the data produces k_{eff} results that are in disagreement with other RCA datasets). The TMI-1 data disagreement illustrates the difficulty of obtaining high quality RCA data, especially for fission products and minor actinides.

Japan may have additional RCA data with fission products. This would be consistent with their laboratory critical experiments with fission products to be used for the Rokkasho reprocessing plant.

Conclusion: RCA data with fission products is the preferred solution, but the TMI-1 data discrepancies must be avoided.

6.4.2 LCE Data

LCEs are the most direct means of solving the fission product cross section data issues for Full Burnup Credit. Several datapoints for a single fission product (rhodium) were obtained by SNL using fresh fuel rods with fission product foils inserted between the pellets in the stack. These experiments were very expensive, but were successful and the Yucca Mountain Project estimated that $54M would provide sufficient data to support all 15 of the Principal Isotope fission products (the 16th is ^{133}Cs). JAEA in Japan published a paper recently that documented their laboratory critical experiments for four fission products simulating the environment of a dissolver at a reprocessing plant. These data appear to be very consistent with calculations indicating a high quality experiment.

Several important fission products do not have covariance matrices in ENDF. This issue was addressed in the development of ENDF/B-VII to some extent, and also independently by ORNL. The covariance matrices are an essential part of the TSUNAMI computer code system used to evaluate the adequacy of benchmark critical experiments.

Conclusion: Laboratory critical experiments would solve the data needs for Full Burnup Credit for cross section validation. However, the costs of acquiring the data may make this solution impractical.

6.4.3 Fission Product Cross Section Measurements

Cross section data may be measured directly by use of a pure sample of a fission product and neutron beams of different energies. Experiments of this type are performed to add to the ENDF data libraries and require a lengthy process of evaluation and review. This type of data could be very valuable for isotopes that have high uncertainties (three fission product isotopes in particular have much higher uncertainties than others). Fission Product isotopes have large thermal cross sections and also have significant resonances. Additional cross section measurements, along with the uncertainties of the measurements, could contribute to improvements in the Indirect Perturbation calculations for cross sections performed by TSUNAMI.

Additional measurements of the major actinides would have little effect, given the large number of measurements that are used to generate the ENDF/B-VII cross section evaluations. Thus, only fission product or minor actinide (Am, Cm) isotopes would be measured as part of a cross section measurement program for Full Burnup Credit.

6.4.4 Database of Reactor Operation History

The existing CRC database could be extended but it is not clear how much benefit would be obtained from this effort. The CRC dataset provides several key pieces of information for Full Burnup Credit and additional data could be helpful. The depletion parameters that apply to the fuel assemblies of a CRC could be evaluated to determine a reasonable range for each depletion parameter.

6.4.5 Axial Burnup Profiles

The existing analyses of Axial Burnup Profiles are adequate to support Full Burnup Credit.

6.4.6 Isotopic Depletion Parameters

The existing studies of depletion parameters are adequate to support Full Burnup Credit, and license applications can avail themselves of proprietary fuel irradiation data if needed.

7 The PIRT Process Applied

A PIRT (Phenomena Identification and Ranking Table) exercise is a formal expert elicitation process with the final output being the ranking tables. The U.S. NRC has utilized the PIRT approach for a range of applications. This project will provide the U.S. DOE with an identification and ranking of the physical processes which determine the credit given for the burnup of spent nuclear fuel.

The goal of the PIRT exercise is to develop input to the DOE for consideration in their efforts to prioritize future development of the experimental data that provide support for the validation of Full Fission Product Burnup Credit. In particular, this PIRT process provides insights into those data areas that are: (1) important, (2) poorly understood or poorly dealt with given the current state of the art, and (3) amenable to additional development.

7.1 Selection of Panelists

Members of the PIRT panel were identified as experienced in the application of burnup credit. The selected panelists span the range that includes the application of burnup credit to spent fuel transport/storage/and disposal. They have participated in the development of ANSI/ANS standards for burnup credit. The panel members were Dr. Dominic Napolitano and Dr. Alan Wells.

Table 7-1: PIRT Definitions

Descriptor:	Definition:
High (H)	First order importance to figure of merit of interest.
Medium (M)	Secondary importance to figure of merit of interest
Low (L)	Negligible importance to figure of merit of interest. Not necessary to model this parameter for this application.
Uncertain (U)	Potentially important. Importance should be explored through sensitivity study and/or discovery experiments and the PIRT revised accordingly

.
The first step in the PIRT process was to rank the phenomena for importance. The importance of the key parameters was relatively easily established. The next stage of the assessment was to rank the state of knowledge with respect to the general adequacy of existing data and methodologies.

Table 7-2: Data Adequacy Descriptors for Existing Model Input and Validation

Descriptor:	Definition:
High (H)	A high resolution database (e.g., validation grade data set) exists, or a highly reliable assessment can be made based on existing knowledge. Data needed are readily available.
Medium (M)	Existing database is of moderate resolution, or not recently updated. Data are available but are not ideal due to age or questions of fidelity. Moderately reliable assessments of models can be made based on existing knowledge.
Low (L)	No existing database or low-resolution database in existence. Assessments cannot be made with even moderate reliability based on existing knowledge.

The final aspect of the model and data adequacy assessment was to rank the feasibility of getting new validation data if the existing data were ranked as anything other than "high" adequacy. The descriptors used for this aspect of the assessment are defined in Table 7-3.

Table 7-3: Potential for Improvement With New Data or Model

Descriptor:	Definition:
High (H)	New Data or Model could provide a substantial improvement.
Medium (M)	New Data or Model could provide a meaningful improvement.
Low (L)	New Data or Model could provide a small improvement.

This last aspect of the PIRT process, the feasibility of getting new input and validation data, was intended to provide an added level of input to the DOE. In particular, the feasibility question was intended to identify the "low hanging fruit" as compared to those aspects of Full Burnup Credit that, while potentially important, might be quite difficult and/or quite expensive to pursue. The rankings relative to feasibility also reflect the technical risk associated with pursuing various aspects of Full Burnup Credit in the future. That is, those items rank high in terms of feasibility should represent low risk undertakings with a high probability of success. In contrast, items ranked with low feasibility will be higher risk undertakings with a significant chance of difficulties with regard to cost or acceptability to the regulator.

7.2 Results of PIRT Evaluation of Full Burnup Credit

The results of the PIRT Evaluation for Full Burnup Credit are tabulated in Table 7-4 and discussed below.

Table 7-4: Results of PIRT Evaluation

Physical Phenomenon	Importance Ranking	Data Adequacy for Input and/or Validation	Potential for Improvement with New Data or Model	Potential Cost
ACTINIDE				
Cross Section Validation[1]	H	H	H (TSUNAMI/ENDF/B-VII)	M
Isotopic Validation[2]	H	H	H (TRITON/ENDF/B-VII)	L
Cross Section Measurement	L	H	M	H
Isotopic (RCA) Measurement	H	H	M	H
FISSION PRODUCT				
Cross Section Validation	H	M	H	M
Isotopic Validation				
Fission Yield	M	H	M	M
Transmutation	H	M	H	H
Radioactive Decay	L	H	L	L
Cross Section Measurement	H	L	H	H
Isotopic Measurement	H	L	H	H
CRCs	M	H	M	M
Isotopic Depletion Parameters (Input)	M	H	M	L
Axial Burnup Profiles (Input)	M	H	L	L

Notes: 1-The potential for improvement for actinide cross section validation comes from the benefit of full integration of the new ENDF/B-VII library into MCNP, KENO, TSUNAMI and other code systems.

2-The potential for improvement for actinide isotopic validation comes from the benefit of full integration of the new ENDF/B-VII library into TRITON.

7.2.1 Explanation of Actinide Rankings

Both the cross section and isotopic data components of Actinide-Only Burnup Credit have very high importance since the biggest effect upon k_{eff} is the depletion of fissile major actinides. The work of many organizations has contributed to the adequacy of the data for the major actinide cross sections and isotopics, and the data adequacy is high. The only future effort envisioned by the panel for these data areas is the full incorporation of ENDF/B-VII into the MCNP, KENO, TSUNAMI, and TRITON code systems. This work has moderate cost due to the complexity of the data preparation and verification efforts, but has high benefit because the end result would be the best quality and best validated Actinide-Only datasets to date.

The importance of cross section measurements for Actinide-Only Burnup Credit is low only because of the continuous efforts of researchers in measuring the cross sections of major actinides. Additional data would be of value to the overall quality of ENDF/B versions, but would not add directly to Actinide-Only Burnup Credit because the current data requires only a relatively small (Δk_{eff} = 0.008) bias that is consistent and well-documented. Considerable additional data might be required to substantially reduce the bias, but since it is added to the final value of k_{eff} anyway, there is no effect for Burnup Credit.

The importance of RCA data has high value because of the on-going changes in LWR fuel assembly complexity. The current dataset focuses mainly on PWR spent fuel, although some data has been acquired for BWR spent fuel. As the enrichments of both types reaches 5.0 wt% (perhaps higher eventually) and burnups increase, additional RCA data is needed to encompass the full range of operational enrichments and burnups. The moderate value expected for additional data is due to the difficulties of adequately defining the irradiation conditions of a single fuel pellet throughout its life in a reactor core. This issue continues to introduce variability into the isotopic measurements so that a reduction in bias and uncertainty is unlikely. Thus, the primary benefit is the expansion of the validation envelope.

7.2.2 Explanation of Fission Product Rankings

The importance of cross section validation and isotopic validation are both high because of the scarcity of fission product data (the data adequacy is Low), and both could benefit greatly from the acquisition of RCA and LCE data. Past measurements in the U.S. have focused on the actinides, since these are the strongest factors in reactor reactivity, and fission products such as xenon, which cannot be credited for burnup credit because it is a gas. The cost of new fission product data acquisition would be high, and measurements face technical challenges due to the difficulties of measuring the relatively small contributions of each individual fission product. LCEs would likely require increases in the relative contents of fission products relative to the uranium isotopes in order to accurately measure the worth of the fission product. RCAs face difficulties that stem from the difficulty of accurately representing the depletion parameters (power level and history of a single pellet), problems that affect actinide RCA measurements as well, plus the difficulty of measuring the small contributions of particular isotopes. These factors contribute to the high costs of fission product data.

7.2.3 Explanation of Ranking for CRCs

The CRC data appears to be adequate for the support of burnup credit validation if it is not the principal means of validation. Improvements might be possible in the definition of the fuel depletion environment of the fuel assemblies, but the cost for this effort could be substantial (M) because investigation by utilities or reactor vendors would be required. The principal reason for pursuing development of new data points would be to reduce the statistical variation due to the relatively small number of datapoints (44), but this could be a worthwhile effort.

7.2.4 Explanation of Depletion Parameter Rankings

The preparers of this report have assigned a Medium importance to isotopic depletion parameters because of the presence of extensive published studies of depletion parameter effects. However, the studies do not consider the simultaneous change of parameters, and changing one parameter in a conservative direction may mandate a change of another in a non-conservative direction.

Current usage of the studies is to assign all parameters a "worst case" value, adding non-physical conservatism. Additional guidance for users regarding a more balanced approach to parameter selection would be beneficial. Further, the studies do not establish upper bounds for the various parameters, and it is left to the user to do this. Since the user may have access to utility data that does establish upper bounds for the depletion parameters, this may not be a serious issue but it still might merit further investigation.

7.2.5 Axial Burnup Profiles

The preparers of this report have assigned Medium importance to this data because the existing published data is comprehensive. Users have a choice of an ORNL guidance document or a statistical approach developed by the YMP, or they may use utility or fuel vendor data to develop their own profiles. The addition of additional data to the axial profile database would be desirable if new fuel types, e.g. MOX fuel or enrichments in excess of 5.0 wt% ^{235}U, were to become necessary.

8 Cost and Schedule Estimates

ENERCON has estimated the costs and schedule for the two options to acquire the necessary data for the validation of Full Burnup Credit. The estimates for the acquisition of traditional RCA and LCE data are based in part upon previous estimates made for the YMP, updated with the current understanding of the validation issues.

The estimates for the updating and improvement (e.g. replacement of the GRESS computer code with a new code based upon the current knowledge of the S/U methodology) of the various parts of the S/U methodology are also based on ENERCON's current understanding of the methodology. ENERCON understands that the efforts at ORNL, especially with regard to isotopic calculations, are still in the exploratory stages. Thus, a cohesive program plan with budget and schedule does not yet exist.

8.1 Estimates for Traditional RCA and LCE Data Acquisition

The cost of acquiring sufficient RCA and LCE data for fission product validation has been estimated previously by the YMP. These estimates have been increased due to a current perception of increased complexity in experimental conditions. The previous ^{103}Rh LCE experimental design should be modified to incorporate the results of a TSUNAMI evaluation that showed that the foils of fission product material were too thick and a new set of diluted foils could improve the results. A readiness review would have to be performed before performing any additional LCEs at SNL. The experimental design guided by TSUNAMI and the readiness review could require a six-month period and $0.5M. The actual fission product LCE effort would require five or more years and cost approximately $15M.

The discrepancies shown in the TMI-1 RCA data highlight the difficulty of fission product isotopic measurements from fuel pellet samples. The environment to which a given fuel pellet was exposed during its time in the reactor is difficult to define with the precision needed for accurate validation results. Thus, an effort to define the exact irradiation conditions prior to the

RCA measurements would be needed. Also, the fission product data obtained for the Limerick BWR power plant showed that the same difficulty affects BWR spent fuel, with even greater complexity due to the presence of water rods, part-length rods, radial and axial enrichment zoning, and gadolinia in some of the fuel rods. The costs of good RCA measurements are difficult to estimate since past experiments have not been very satisfactory, and it would seem that more data would be required than expected by the YMP to insure that the complexity of the validation is properly addressed. The preferred method to do this is to divide each sample in thirds and have three independent laboratories perform the RCA analyses. The costs of shipment remain a large part of the cost of RCAs, and shipment to three different laboratories would help triple the previously expected costs. Thus the costs could range from $30M to almost three times that, but economies of scale for such a large effort could mitigate the overall costs of the RCAs.

Total costs of the RCA and LCE data acquisition are estimated by ENERCON to be $50M to $100M and require five years or more. Costs of measurements of fission product cross sections are not included.

8.2 Estimates for S/U Isotopic and Cross Section Development

The costs and schedule for S/U refinement (addition of ENDF/B-VII data) and development of a replacement to the GRESS computer code system resemble a computer programming project, since the work relies heavily on computer software. The nuclear data needed by the various software components is available in ENDF/B-VII and other references. The possibility of improving the covariance matrices for fission products that are contained in ENDF/B-VII has been considered. The covariance matrices are a form of uncertainty data that forms the basis for most of the S/U calculations. This would be a large effort requiring an undetermined number of man-years at ORNL, and it does not seem likely that such work could be completed in a two-three year time period. It would be possible to revise the covariance matrices separately with the intent of developing an improved Full Burnup Credit at a later date.

The S/U cross section validation effort is very mature and is nearly in final form (accepting the current fission product covariance matrices as is). This is the result of the extensive development of the methodology and the use of the computer codes by DOE sites to address criticality validation issues for materials such as ^{233}U. Preparation of the necessary technical basis documents for the NRC appears to be feasible within a several year period. An effort could be made to find ways to use the data from the international community and the CRCs as confirmatory data.

The S/U isotopic validation effort is not mature, although it was explored by EPRI as mentioned above. The biggest part of the effort will be to replace the GRESS computer code system, since it is based upon the obsolete SAS2H code system no longer supported as part of SCALE. The level of effort for this code development is about two man-years. A scoping analysis would be performed prior to code development to resolve the nuclear technical details, which would require a half-man-year. It is not clear if additional data is necessary for the fission yield portion of the effort, but the possibility was discussed.

The total costs would be for three to five dedicated researchers for three years with support personnel as necessary, and ENERCON estimates a cost of $5M total and a schedule of 2-3 years time for the effort. The potential for revisions to the ENDF/B-VII covariance matrices and the fission yield uncertainty data, plus the possibility of unforeseen difficulties, could increase this estimate to $10M.

8.3 Organization Responsibilities

The research efforts for data acquisition and development of an advanced form of the S/U methodology for Full Burnup Credit are best handled by the National Laboratories, including ORNL for S/U methodologies, SNL for fission product LCEs, and ORNL or General Electric's Vallecitos for RCAs. Data gathering for CRCs, depletion parameters, and axial burnup profiles could be handled by commercial nuclear industry entities with ties to reactor sites. Coordination between the National Laboratories and reactor sites could be provided by commercial nuclear industry entities.

9 Conclusions

A PIRT analysis was performed by ENERCON to evaluate the processes of acquiring nuclear data which determine burnup credit to support industry needs, and to evaluate the processes relative to each other. The PIRT analysis produced several results:

- Additional experimental data to support the Actinide-Only burnup credit is not necessary. The existing RCA data plus the French HTC data plus the LCE data provide adequate benchmark validation.
- The cost and schedule for additional experimental RCA and LCE data to support the addition of 16 fission products to produce Full Burnup Credit are relatively high.
- Another process for obtaining fission product burnup credit is the S/U analysis method being developed by ORNL and EPRI.
- The cost and schedule for the S/U analysis approach is significantly less that for obtaining additional experimental data.

Thus, ENERCON concludes that there are two options for moving forward with the addition of 16 Fission Products to Actinide-Only Burnup Credit to obtain Full Burnup Credit. These options are discussed below with an ENERCON estimate of schedule and cost in addition to identifying the organization best suited to perform the task.

Option A: Experimental Data Based Method

The current state of the art for RCAs and LCEs internationally could provide a source of additional data for Full Burnup Credit. The acquisition of similar data within the US is possible with LCEs at SNL and isotopic measurements at a national or commercial laboratory and cross section measurements at a variety of facilities, although at a substantial cost. The biggest advantage of developing new data within the US would be that the statistical data and analysis necessary to support the NRC review and approval could be achieved with a cohesive data program. The preliminary estimate by ENERCON of the cost could be as high as $100M.

Organizations:	National Laboratories or Commercial Facilities
Schedule:	5 years or more, depending on level of funding
Estimated Cost:	$50M to $100M

Option B: Sensitivity/Uncertainty (S/U) Method

A more cost-effective approach would be to focus on improvements in the definition of existing actinide RCAs, TSUNAMI cross section uncertainty analyses, and evaluations of isotopic uncertainties by extension of cross section uncertainties with the GRESS code (or improved code) plus evaluations of the fission yield and radioactive decay. ENERCON believes that this S/U method could be accomplished on a much shorter schedule than the acquisition of new experimental data and the preliminary costs are estimated by ENERCON to be in the range of $10M.

Organization:	ORNL
Schedule:	2-3 years, depending on level of funding
Estimated Cost:	$5 to10 M

In addition, the newest version of the nuclear dataset, ENDF/B-VII, can be incorporated into the traditional and S/U computer analyses used in the validation of Full Burnup Credit. Also, the CRC data can also be revisited as confirmatory data for the isotopic bias term. New LCE or RCA data on fission products can be incorporated as it becomes available.

ENERCON believes that the data activities described in the two options are not exclusive of each other. For example, the S/U method may be pursued while experimental data gathering is also pursued. The timeframe for data gathering through experiments tends to be much longer than the timeframe for calculations with the S/U method (2-3 years). Thus, any experimental data incorporated into the Option B would be from programs already in progress. Incorporation of new experimental measurements into Option B would compromise the 2-3 year schedule estimated by ENERCON.

10 Recommendations

ENERCON recommends that the S/U analysis method should be developed in more detail and presented to the NRC for their review and consideration as a method for adding fission products to the Actinide-Only Burnup Credit. The reasoning behind this recommendation is summarized below:

- ENERCON believes the nuclear industry wants to improve upon Actinide-Only Burnup Credit as soon as possible. Actinide-Only Burnup Credit is limiting the benefit of burnup credit, resulting in compromises in used nuclear fuel transportation and storage cask design and operational loading curves for casks, and increases program costs and personnel exposures.

- The value of the S/U methodology as a solution to attaining Full Burnup Credit has been identified in this report but must be confirmed by interactions with the NRC.
- Resolution of the validation issues that have hampered efforts to obtain Full Burnup Credit is possible using the S/U methodology with existing data. The existing data (e.g. the covariance matrices), should be sufficient to allow approval of some form of Full Burnup Credit without the performance of additional experiments.
- The near-term potential for a positive response from the NRC for some form of fission product burnup credit could clarify the data needs for long-term improvements in the amount of credit given.

ENERCON concludes that the nuclear industry would prefer to have Full Burnup Credit available as soon as it can be achieved but will take advantage of any improvement upon Actinide-Only Burnup Credit as soon as it is available. ENERCON believes that the S/U methodology provides the quickest path to improving Actinide-Only Burnup Credit. ENERCON understands that the existing experimental data may not be sufficient to support the S/U methodology to gain approval for 100 percent credit for each of the 16 fission product and 4 minor actinides. If this is the case, ENERCON recommends that Full Burnup Credit with de-rated credit for some of the fission products be pursued. Further improvements of Full Burnup Credit could be expected in the future as additional and improved experimental data become available.

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12 Bibliography

Action item from April 24 meeting on French experimental data, Cecil Parks, Group Leader, Nuclear Analysis Methods and Applications, Oak Ridge National Laboratory, May 23, 2002

"Attached are letter reports prepared for the Nuclear Regulatory Commission to 1) provide our recommendations on the acquisition of experimental data from the French program on burnup credit and 2) document the quantitative assessment performed on open literature and proprietary experimental data relevant to burnup credit. The quantitative assessment was performed using sensitivity/uncertainty methods we have been working to develop and use here at Oak Ridge National Laboratory. Note the quantitative assessment was performed subsequent to our initial recommendations."

Context of the Burn-Up Credit, COGEMA

"Negative reactivity of actinides and FP in pcm (PWR 17x17, Cooling time = 5 ans)"

Dry Storage of High-Burnup Spent Fuel – Status of Interactions with NRC, Albert J. Machiels, Spent Fuel Steering Committee Meeting, Charlotte, NC, August 21, 2001

"…the spent fuel cladding must be protected during storage *against degradation that leads to gross ruptures* or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems . . . {CFR, Title 10, Part 72.122(h) (1)}"

Robust Fuel Program Hot Cell Examination Projects, Odelli Ozer, EPRI Project Manager, EPRI, Spent Fuel Storage, Transportation and Disposal Target Meeting, Charlotte, NC, August 21, 2001

"Objectives of the Robust Fuel Program"

DOE and NRC Spent Fuel Storage/Disposal Research, H.B. Robinson & Limerick Station Fuel

"ANL Fuel Rod Characterization"

Comparison of French and Yucca Mountain Fission Product Approaches

"Criticality safety for storage and transportation casks has traditionally depended upon the "fresh fuel" assumption, which treats all irradiated spent nuclear fuel (SNF) as if it retained all of the initial U-235 enrichment. The depletion of the U-235 fissile isotope dramatically reduces the neutronic reactivity of spent fuel, although the build-in of plutonium isotopes compensates for some of the reactivity reduction. . . ."

HTC rods, HTC experiments, PART 5: Mixed HTC/UO_2 rods, arrays, etc.

"UO_2 PUO_2 . . . Composition of U (4.5) O_2 rod irradiated until 37 500 $MWdt^{-1}$ without fission products"

HTC Programme, High Burnup Rods Array for Criticality Studies, General Presentation Parts 2 to 4, SEC/T/01.240 DRAFT, Institut de Protection et de Surete Nucleaire (IPSN), Departement de Prevention et d'Etude des Accidents, Service d'Etudes de Criticite

"RESUME: Le document comprend une presentation en Anglais des quatres premieres phases du programme experimental HTC (Haut Taux de Combustion)."

Burn-up Credit Experimental Programs, COGEMA, Velizy, le 06/08/2001

"Objet: Les Programmes Experimentaux du credit burn-up"

Burn-Up Credit and associated measurements, Doc/0083/12/SQY/00.0000.Rev. 0, COGEMA

"Current application of the actinide-only Burn-Up Credit in the COGEMA Group and R&D programme to take fission products into account"

Current Applications of Actinide-only Burn-up Credit within the COGEMA Group and R&D Programme to take Fission Products into Account, H. Toubon & Guillou of COGEMA, P. Cousinou and F. Barbry of Institute for Nuclear Protection and Safety, J.P. Grouiller & G. Bignan of CEA Cadarache, ICONE 9, Nice, France, April 2001

"**Burn-up credit background and context -** Burn-up credit can be defined as making allowance for absorbent radioactive isotopes in criticality studies, in order to optimise safety margins and avoid over-engineering of nuclear facilities."

Burnup Credit for Fission Product Nuclides in PWR (UO_2) Spent Fuels, N. Thiollay, J.P. Chauvin, B. Roque, A. Santamarina, J. Pavageau, J.P. Hudelot, H. Toubon, CEA – Cadarache DRN/DER, COGEMA/DSCP/DSD/SDU, ICNC 99, September 20-24, 1999, Versailles, France

"The extensive experimental programme developed at CEA-Cadarache, to allow reactivity burnup credit in PWR spent fuel calculations, is described. Chemical analysis of PWR pins supplied the fuel inventory, i.e., Actinides and Fission Products. Reactivity worth measurements of the various burnup credit nuclides were performed in the MINERVE reactor by an oscillation technique. The Calculation-Experiement analyses based on French criticality-safety calculation packages (the current package CRIBLE and the next package CRISTAL) are presented."

Burnup Credit in LWR-MOX Assemblies, B. Roque, A. Santamarina, N. Thiollay, CEA – Cadarache, DRN/DER/SPRC

"The current increase of French MOX fuel cycle enhances the need for criticality-safety design studies linked to plutonium-fueled assemblies. Therefore, the Burnup Credit in MOX fuel is an important challenge in France and is investigated in this paper."

Contribution to the Experimental Validation of the New French Criticality-Safety Package 'CRISTAL', B. Roque, A. Santamarina, C. Mattera, CEA – Cadarache, DRN/DER/SPRC and E. Lejeune, SGN – Direction Technique, ICNC 99, September 20-24, 1999, Versailles, France

"In order to repond to the increasing accuracy requirements for French fuel cycle applications, the Commissariat a l'Energie Atomique (CEA), the Institut de Protection et de Surete Nuceaire (IPSN) and the COGEMA/SGN Company, are developing a new criticality-safety package 'CRISTAL'. The CRISTAL system involves the newest accurate codes, APOLLO2, MORET4, and TRIPOLI4 and uses the recent JEF2 European File. The design calculation routes were elaborated and checked by reference code calculations. Then, these recommended routes were validated against specific Criticality experiments."

Contribution To The Validation Of Burn-up Credit In French UOX and MOX Spent Fuels, N. Thiollay, B. Roque, J. Hudelot, H. Toubon, CEA-Cadarache, DRN/DER, COGEMA, BU-T/DPP/SRD, Nuclear Data for Sciences & Technology, Tsukuba, Japan, 7-12/10/01

"A large experimental program has been performed within a CEA and COGEMA cooperation to validate burn up credit for fission products in various types of fuels and implement it in the French spent fuel management. Their nuclear data have been investigated throughout two kinds of experiments: . . ."

Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data, NUREG/CR-6979 ORNL/TM-2007/083, D.E. Mueller, K.R. Elam, P.B. Fox, September 2008

"In the 1980s, a series of critical experiments referred to as the Haut Taux de Combustion (HTC) experiments was conducted by the Institut de Radioprotection et de Surete Nucleaire (IRSN) at the experimental criticality facility in Valduc, France. The plutonium-to-uranium ratio and the isotopic compositions of both the uranium and plutonium used in the simulated fuel rods were designed to be similar to what would be found in a typical pressurized-water reactor fuel assembly that initially had an enrichment of 4.5 wt % ^{235}U 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 HTC experiments include configurations designed to simulate fuel handling activities, pool storage, and transport in casks constructed of thick lead or steel. . . "

<u>Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel</u>, NUREG/CR-6665 ORNL/TM-1999/303, C.V. Parks, M.D. DeHart, J.C. Wagner, February 2000

"This report has been prepared to review relevant background information and provide technical discussion that will help initiate a PIRT (Phenomena Identification and Ranking Tables) process for use of burnup credit in light-water reactor (LWR) spent fuel storage and transport cask applications. The PIRT process will be used by the NRC Office of Nuclear Regulatory Research to help prioritize and guide a coordinated program of research and as a means to obtain input/feedback from industry and other interested parties. The review and discussion in this report are based on knowledge and experience gained from work performed in the United States and other countries. Current regulatory practice and perceived industry needs are also reviewed as a background for prioritizing technical needs that will facilitate safe practice in the use of burnup credit. Relevant physics and analysis phenomenon are identified, and an assessment of their importance to burnup credit implementation is given. Finally, phenomena that need to be better understood for effective licensing, together with technical issues that require resolution, are presented and discussed in the form of a prioritization ranking and initial draft program plan."

Investigation of Burnup Credit Modeling Issues Associated With BWR Fuel, ORNL/TM-1999/193, J.C. Wagner, M.D. DeHart, B.L. Broadhead, October 2000

"This report investigates various calculational modeling issues associated with boiling-water-reactor (BWR) fuel depletion relevant to burnup credit. To date, most of the efforts in burnup-credit studies in the United States have focused on issues related to pressurized-water-reactor (PWR) fuel. However, requirements for the permanent disposal of BWR fuel have necessitated the development of methods for predicting the spent fuel contents for such fuels. Concomitant with such analyses, validation is also necessary. This report provides a summary of initial efforts to better understand and validate away-from-reactor spent fuel analysis methods for BWR fuel. These efforts include: assessment of SAS2H for BWR depletion calculations by code-to-code comparisons with HELIOS, investigation of SAS2H modeling issues and depletion assumptions, and finally, analysis of the sensitivity of three-dimensional criticality calculations to depletion assumptions."

<u>Fission Product Benchmarking for Burnup Credit Applications</u> Progress Report 1002879, Alan H. Wells, EPRI Interim Report, December 2002

"This report presents the progress to date toward developing a technical basis, together with the collection of supporting data, for a cost-effective burnup credit methodology for spent nuclear fuel with initial U-235 enrichment up to 5%."

Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages, Revision 2, DOE/RW-0472 Rev. 2, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, September 1998

"The Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages describes a methodology for performing and applying nuclear criticality safety calculations with actinide-only burnup credit. The changes in the U-234, U-235, U-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241 concentration with burnup are used in burnup credit criticality analyses. No credit for fission product neutron absorbers is taken. The methodology consists of five major steps. . . "

Validation of Scale (SAS2H) Isotopic Predictions for BWR Spent Fuel, O.W. Hermann, M.D. DeHart, ORNL/TM-13315, September 1998

"Thirty spent fuel samples obtained from boiling-water-reactor (BWR) fuel pins have been modeled at Oak Ridge National Laboratory using the SAS2H sequence of the SCALE code system. The SAS2H sequence uses transport methods combined with the depletion and decay capabilities of the ORIGEN-S code to estimate the isotopic composition of fuel as a function of its burnup history. Results of these calculations are compared with chemical assay measurements of spent fuel inventories for each sample. Results show reasonable agreement between measured and predicted isotopic concentrations for important actinides, however, little data are available for most fission products considered to be important for spent fuel concerns (e.g., burnup credit, shielding, source-term calculations, etc.)"

Limited Burnup Credit in Criticality Safety Analysis: A Comparison of ISG-8 and Current International Practice, I.C. Gauld, NUREG/CR-6702, ORNL/TM-2000/72, January 2001

"This report has been prepared to qualitatively assess the amount of burnup credit (reactivity margin) provided by ISG-8 compared to that provided by the burnup credit methodology developed and currently applied in France. For the purposes of this study, the methods proposed in the DOE Topical Report have been applied to the ISG-8 framework since this methodology (or one similar to it) is likely to form the basis of initial cask licensing applications employing limited burnup credit in the United States. This study limits the scope of the comparison to several of the fundamental burnup credit parameters: the nuclides credited in the analysis, the axial-burnup profile, and the cooling time. An investigation of other parameters, such as horizontal burnup effects and isotopic correction factors to account for biases and uncertainties in calculated actinide compositions, were beyond the scope of this review. This report compares the amount of burnup credit provided by the respective methodologies for typical axial-burnup profiles derived from averaging actual PWR axial-burnup distributions. In addition, a limited assessment of several atypical axial-burnup distributions is also included."

Validation of Scale System for PWR Spent Fuel Isotopic Composition Analyses, O.W. Hermann, S.M. Bowman, M.C. Brady, C.V. Parks, ORNL/TM-12667, March 1995

"The validity of the computation of pressurized-water-reactor (PWR) spent fuel isotopic composition by the SCALE system depletion analysis was assessed using data presented in the report. Radiochemical measurements and SCALE/SAS2H computations of depleted fuel isotopics were compared with 19 benchmark-problem samples from Calvert Cliffs Unit 1, H.B. Robinson Unit 2, and Obrigheim PWRs. Even though not exhaustive in scope, the validation included comparison of predicted and measured concentrations for 14 actinides and 37 fission and activation products."

An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel, M.D. DeHart, O.W. Hermann, ORNL/TM-13317, September 1996

"Isotopic characterization of spent fuel via depletion and decay calculations is necessary for determination of source terms for subsequent system analyses involving heat transfer, radiation shielding, isotopic migration, etc. Unlike fresh fuel assumptions typically employed in the criticality safety analysis of spent fuel configurations, burnup credit applications also rely on depletion and decay calculations to predict the isotopic composition of spent fuel. These isotopics are used in subsequent criticality calculations to assess the reduced worth of spent fuel. To validate the codes and data used in depletion approaches, experimental measurements are compared with numerical predictions for relevant spent fuel samples. Such comparisons have been performed in earlier work at the Oak Ridge National Laboratory (ORNL). This report describes additional independent measurements and corresponding calculations, which supplement the results of the earlier work. The current work includes measured isotopic data from 19 spent fuel samples obtained from the Italian Trino Vercelles pressurized-water reactor (PWR) and the U.S. Turkey Point Unit 3 PWR."

Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit, J.C. Wagner, NUREG/CR-6747, ORNL/TM-2000/306, October 2001

"This report proposes and documents a computational benchmark problem for the estimation of the additional reactivity margin available in spent nuclear fuel (SNF) from fission products and minor actinides in a burnup-credit storage/transport environment, relative to SNF compositions containing only the major actinides. The benchmark problem/configuration is a generic burnup credit cask designed to hold 32 pressurized water reactor (PWR) assemblies. The purpose of this computational benchmark is to provide a reference configuration for the estimation of the additional reactivity margin, which is encouraged in the U.S. Nuclear Regulatory Commission (NRC) guidance for partial burnup credit (ISG8), and document reference estimations of the additional reactivity margin as a function of initial enrichment, burnup, and cooling time. Consequently, the

geometry and material specifications are provided in sufficient detail to enable independent evaluations. Estimates of additional reactivity margin for this reference configuration may be compared to those of similar burnup-credit casks to provide an indication of the validity of design-specific estimates of fission-product margin. The reference solutions were generated with the SAS2H-depletion and CSAS25 sequences have been extensively validated elsewhere, the reference solutions are not directly or indirectly based on experimental results. Consequently, this computational benchmark cannot be used to satisfy the ANS 8.1 requirements for validation of calculational methods and is not intended to be used to establish biases for burnup credit analyses."

Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel, L.C. Gauld, C.V. Parks, NUREG/CR-6701, ORNL/TM-2000/277, January 2001

"This report has been prepared to review the technical issues important to the prediction of isotopic compositions and source terms for high-burnup, light-water-reactor (LWR) fuel as utilized in the licensing of spent fuel transport and storage systems. The current trend towards higher initial ^{235}U enrichments, more complex assembly designs, and more efficient fuel management schemes has resulted in higher spent fuel burnups than seen in the past. This trend has led to a situation where high-burnup assemblies from operating LWRs now extend beyond the area where available experimental data can be used to validate the computational methods employed to calculate spent fuel inventories and source terms. This report provides a brief review of currently available validation data, including isotopic assays, decay heat measurements, and shielded dose-rate measurements. Potential new sources of experimental data available in the near term are identified. A review of the background issues important to isotopic predictions and some of the perceived technical challenges that high-burnup fuel presents to the current computational methods are discussed. Based on the review, the phenomena that need to be investigated further and the technical issues that require resolution are presented. The methods and data development that may be required to address the possible shortcomings of physics and depletion methods in the high-burnup and high-enrichment regime are also discussed. Finally, a sensitivity analysis methodology is presented. This methodology is currently being investigated at Oak Ridge National Laboratory as a computational tool to better understand the changing relative significance of the underlying nuclear data in the different enrichment and burnup regimes and to identify the processes that are dominant in the high-burnup regime. The potential application of the sensitivity analysis methodology to help establish a range of applicability for experimental data in code validation is also discussed and demonstrated."

Assessment of Fission Product Cross-Section Data for Burnup Credit Applications, L.C. Leal, H. Derrien, M.E. Dunn, D.E. Mueller, ORNL/TM-2005/65, December 2007

"Past efforts by the Department of Energy (DOE), the Electric Power Research Institute (EPRI), the Nuclear Regulatory Commission (NRC), and others have provided sufficent technical information to enable the NRC to issue regulatory guidance for implementation of pressurized-water reactor (PWR) burnup credit; however, consideration of only the reactivity change due to the major actinides is recommended in the guidance. Moreover, DOE, NRC, and EPRI have noted the need for additional scientific and technical data to justify expanding PWR burnup credit to include fission product (FP) nuclides and enable burnup credit implementation for boiling-water reactor (BWR) spent nuclear fuel (SNF). The criticality safety assessment needed for burnup credit applications will utilize computational analyses of packages containing NSF with FP nuclides. Over the years, significant efforts have been devoted to the nuclear data evaluation of major isotopes pertinent to reactor applications (i.e., uranium, plutonium, etc.); however, efforts to evaluate FP cross-section data in the resonance region have been less thorough relative to actinide data. In particular, resonance region cross-section measurements with corresponding R-matrix resonance analyses have not been performed for FP nuclides. Therefore, the objective of this work is to assess the status and performance of existing FP cross-section and cross-section uncertainty data in the resonance region for use in burnup credit analyses. Recommendations for assessment focuses on seven primary FP isotopes (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{151}Sm, ^{152}Sm, and ^{155}Gd) that impact reactivity analyses of transportation packages and two FP isotopes (^{153}Eu and ^{155}Eu) that impact prediction of ^{155}Gd concentrations . . . . . "

<u>Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks</u>, S.M. Bowman, I.C. Gauld, J.C. Wagner, NUREG/CR-6716, ORNL/TM-2000/385, March 2001

"The U.S. Nuclear Regulatory Commission (NRC) is currently reviewing the technical specifications for spent fuel storage casks in an effort to develop standard technical specifications (STS) that define the allowable spent nuclear fuel (SNF) contents. One of the objectives of the review is to minimize the level of detail in the STS that define the acceptable fuel types. To support this initiative, this study has been performed to identify potential fuel specification parameters needed for criticality safety and radiation shielding analysis and rank their importance relative to a potential compromise of the margin of safety."

<u>Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit</u>, G. Radulescu, D.E. Mueller, J.C. Wagner, NUREG/CR-6951, ORNL/TM-2006/87, December 2007

"The purpose of this study is to provide insights into the neutronic similarities that may exist between a generic cask containing typical spent nuclear fuel assemblies and commercial reactor critical (CRC) state-points. Forty CRC state-points from five pressurized-water reactors were selected for the study and the type of CRC state-points that may be applicable for validation of burnup credit criticality safety calculations for spent fuel transport/storage/disposal systems are identified. The study employed cross-section sensitivity and uncertainty analysis methods developed at Oak Ridge National Laboratory and the TSUNAMI set of tools in the SCALE code system as a means to investigate system similarity on an integral and nuclide-reaction specific level. The results indicate that, except for the fresh fuel core configuration, all analyzed CRC state-points are either highly similar, similar, or marginally similar to a generic cask containing spent nuclear fuel assemblies with burnups ranging from 10 to 60 GWd/MTU. . . . "

<u>Feasibility of Direct Disposal of Dual-Purpose Canisters, Options for Assuring Criticality Control</u>, Alan H. Wells, EPRI 1016629, Final Report, December 2008

"This report presents calculated nuclear reactivities of two dual-purpose spent-fuel canisters to assess the feasibility, with respect to criticality control, of direct disposal of such canisters in a permanent geologic repository without the need for repackaging. Results show that criticality safety cannot always be unequivocally demonstrated through burnup credit alone, except by taking into account a sufficiently large number of neutron-absorbing fission products and reasonable values for biases and uncertainties. However, criticality control for direct disposal can be significantly enhanced with the inclusion of used burnable absorber rods in spent-fuel assemblies and loading patterns that minimize reactivity."

<u>Evaluation of Cross-Section Sensitivities in Computing Burnup Credit Fission Product Concentrations</u>, I.C. Gauld, D.E. Mueller, ORNL/TM-2005/48, August 2005

"U.S. Nuclear Regulatory Commission Interim Staff Guidance 8 (ISG-8) for burnup credit covers actinides only, a position based primarily on the lack of definitive critical experiments and adequate radiochemical assay data that can be used to quantify the uncertainty associated with fission product credit. The accuracy of fission product neutron cross sections is paramount to the accuracy of criticality analyses that credit fission products in two respects: (1) the microscopic cross sections determine the reactivity worth of the fission products in spent fuel and (2) the cross sections determine the reaction rates during irradiation and thus influence the accuracy of predicted final concentrations of the fission products in the spent fuel. This report evaluates and quantifies the importance of the fission product cross sections in predicting concentrations of fission products proposed for the use in burnup credit. The study includes an assessment of the major fission products in burnup credit and their production precursors. Finally, the cross-section importance, or sensitivities, are combined with the importance of each major fission product to the system eigenvalue (k_{eff}) to determine the net importance of cross sections to k_{eff}. The importances established the following fission products, listed in descending order of priority, that are most likely to benefit burnup credit when their cross-section uncertainties are reduced: ^{151}Sm, ^{103}Rh, ^{155}Eu, ^{150}Sm, ^{152}Sm, ^{153}Eu, and ^{143}Nd."

Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs, J.C. Wagner, C.E. Sanders, NUREG/CR-6800, ORNL/TM-2002/6, March 2003

"This report presents studies to assess reactivity margins and loading curves for pressurized water reactor (PWR) burnup-credit criticality safety evaluations. The studies are based on a generic high-density 32-assembly cask and systematically vary individual calculational (depletion and criticality) assumptions to demonstrate the impact on the predicted effective neutron multiplication factor, k_{eff}, and burnup-credit loading curves. The purpose of this report is to provide a greater understanding of the importance of input parameter variations and quantify the impact of calculational assumptions on the outcome of a burnup-credit evaluation. This study should provide guidance to regulators and industry on the technical areas where improved information will most enhance the estimation of accurate sub-critical margins. Based on these studies, areas where future work may provide the most benefit are identified. The report also includes an evaluation of the degree of burnup credit needed for high-density casks to transport the current spent nuclear fuel inventory. By comparing PWR discharge data to actinide-only based loading curves and determining the number of assemblies that meet the loading criteria, this evaluation finds that additional negative reactivity (through either increased credit for fuel burnup or cask design/utilization modifications) is necessary to accommodate the majority of current spent fuel assemblies in high-capacity casks. Assemblies that are not acceptable for loading in the prototypic high-capacity cask may be stored or transported by other means (e.g., lower capacity casks that utilize flux traps and/or increased fixed poison concentrations or high-capacity casks with design/utilization modifications)."

STARBUCS: A Prototypic SCALE Control Module for Automated Criticality Safety Analyses Using Burnup Credit, I.C. Gauld, S.M. Bowman, NUREG/CR-6748, ORNL/TM-2001/33, October 2001

"STARBUCS is a new prototypic analysis sequence for performing automated criticality safety analyses of spent fuel systems employing burnup credit. A depletion analysis calculation for each of the burnup=dependent regions of a spent fuel assembly, or other system containing spent fuel, is performed using the ORIGEN-ARP sequence of SCALE. The spent fuel compositions are then used to generate resonance self-shielded cross sections for each region of the problem, which are applied in a three-dimensional criticality safety calculation using the KENO V.a code. This prototypic burnup credit analysis sequence allows the user to simulate the axial and horizontal burnup gradients in a spent fuel assembly, select the specific actinides and/or fission products that are to be included in the criticality analysis, and apply isotopic correction factors to the predicted spent fuel nuclide inventory to account for calculational bias and uncertainties. Although STARBUCS was developed to address the burnup credit analysis needs for spent fuel transport and storage applications, it provides sufficient flexibility to allow criticality safety assessments involving many different potential configurations of spent nuclear fuel to be simulated."

Isotopic Model for Commercial SNF Burnup Credit, Uncanistered Spent Nuclear Fuel, CAL-DSU-NU-000007 Rev 00A, Alan H. Wells, Jason E. Huffer, August 2004

"*Disposal Criticality Analysis Methodology Topical Report* describes a methodology for performing post-closure criticality analyses within the repository at Yucca Mountain, Nevada. An important component of the post-closure criticality analysis is the calculation of conservative isotopic concentrations for spent nuclear fuel. This report documents the isotopic calculation methodology. The isotopic calculation methodology is shown to be conservative based upon current data for pressurized water reactor and boiling water reactor spent nuclear fuel. . ."

Investigation of Burnup Credit Modeling Issues Associated with BWR Fuel, J.C. Wagner, M.D. DeHart, B.L. Broadhead, ORNL/TM-1999-193, October 2000

"This report investigates various calculational modeling issues associated with boiling-water-reactor (BWR) fuel depletion relevant to burnup credit. To date, most of the efforts in burnup-credit studies in the United States have focused on issues related to pressurized-water-reactor (PWR) fuel. However, requirements for the permanent disposal of BWR fuel have necessitated the development of methods for predicting the spent fuel contents for such fuels. Concomitant with such analyses, validation is also necessary. This report provides a summary of initial efforts to better understand and validate away-from-reactor spent fuel analysis methods for BWR fuel. These efforts include: assessment of SAS2H for BWR depletion calculations by code-to-code comparisons with HELIOS, investigation of SAS2H modeling issues and depletion assumptions, and finally, analysis of the sensitivity of three-dimensional criticality calculations to depletion assumptions. . . ."

Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor, C.E. Sanders, I.C. Gauld, NUREG/CR-6798, ORNL/TM-2001/259, January 2003

"This report presents the results of computer code benchmark simulations against spent fuel radiochemical assay measurements from the Kansai Electric Ltd. Takahama-3 reactor published by the Japan Atomic Energy Research Institute. Takahama-3 is a pressurized-water reactor that operates with a 17 x 17 fuel-assembly design. Spent fuel samples were obtained from assemblies operated for 2 and 3 cycles and achieved a maximum burnup of 47 GWd/MTU. Radiochemical analyses were performed on two rods having an initial enrichment of 4.22 wt %, and one integral burnable absorber rod containing Gd_2O_3. These measurements represent the highest enrichment and highest burnup samples currently available in the United States. The benchmark results are important to burnup credit initiatives in the United States since the lack of available benchmark

data has led to restrictions on the allowable credit beyond 4.0 wt % and 40 GWd/MTU. Although the primary objective of the measurements was support of burnup credit, radiochemical analyses were also available for a number of actinide and fission product nuclides important to decay heat and radiation source term analysis. Isotopic predictions from both the SCALE 4.4a and HELIO-1.6 code systems were used in this benchmark study. The results indicate that the level of agreement between predictions and measurements is very good. The results, for the most part, are consistent with the findings of earlier studies for lower enrichment and lower burnup samples and yield similar biases and levels of uncertainty."

Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses, J.C. Wagner, M.D. Dehart, C.V. Parks, NUREG/CR-6801, ORNL/TM-2001/273, March 2003

"This report presents studies performed to support the development of a technically justifiable approach for addressing the axial-burnup distribution in pressurized-water reactor (PWR) burnup-credit criticality safety analyses. The effect of the axial-burnup distribution on reactivity and proposed approaches for addressing the axial-burnup distribution are briefly reviewed. A publicly available database of profiles is examined in detail to identify profiles that maximize the neutron multiplication factor, k_{eff}, assess its adequacy for PWR burnup credit analyses, and investigate the existence of trends with fuel type and/or reactor operations. A statistical evaluation of the k_{eff} values associated with the profiles in the axial-burnup-profile database was performed, and the most reactive (bounding) profiles were identified as statistical outliers. The impact of these bounding profiles on k_{eff} is quantified for a high-density burnup credit cask. Analyses are also presented to quantify the potential reactivity consequence of loading assemblies with axial-burnup profiles that are not bounded by the database. The report concludes with a discussion on the issues for consideration and recommendations for addressing axial burnup in criticality safety analyses using burnup credit for dry cask storage and transportation."

Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation – Calvert Cliffs, Takahama, and Three Mile Island Reactors, G. Ilas, I.C. Gauld, F.C. Difilippo, M.B. Emmett, NUREG/CR-6968, ORNL/TM-2008/071, February 2010

"This report is part of a report series designed to document benchmark-quality radiochemical isotopic assay data against which computer code accuracy can be quantified to establish the uncertainty and bias associated with the code predictions. The experimental data included in the report series were acquired from domestic and international programs and include spent fuel samples that cover a large burnup range. The measurements analyzed in the current report, for which experimental data is publicly available, include 38 spent fuel samples selected from fuel rods with 2.6 to 4.7 wt % ^{235}U initial enrichment, which were irradiated in three pressurized water reactors operated in the United States and Japan and achieved burnup values from 14 to 56 GWd/MTU. The analysis of the measurements was performed by employing the two-dimensional depletion sequence of the TRITON module in the SCALE code system."

Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation – ARIANE and REBUS Programs (UO_2 Fuel), G. Ilas, I.C. Gauld, B.D. Murphy, NUREG/CR-6969, ORNL/TM-2008/072, February 2010

"This report is part of a report series designed to document benchmark-quality radiochemical assay data against which computer code predictions of isotopic composition for spent nuclear fuel can be validated to establish the uncertainty and bias associated with the code predictions. The experimental data analyzed in the present report were acquired from two international programs: (1) ARIANE and (2) REBUS, both coordinated by Belgonnucleaire. All measurements include extensive actinide and fission product data of importance to spent fuel safety applications including burnup credit, decay heat, and radiation source terms. The analyzed four spent fuel samples were selected from fuel rods with initial enrichments of 3.5, 3.8, and 4.1 wt % ^{235}U, which were irradiated in two pressurized water reactors operated in Germany and Switzerland to reach burnups in the 30 to 60 GWd/MTU range. Analysis of the measurements was performed by using the two-dimensional depletion sequence of the TRITON module in the SCALE computer code system."

A Criticality Code Validation Exercise for a LEU Lattice, Bradley T. Rearden, ANS Winter Meeting, Albuquerque, NM, Nover 12-16, 2006

"In the criticality code validation of common systems, many paths to a correct bias, bias uncertainty, and upper subcritical limit may exist. The challenge for the criticality analyst is to select an efficient, defensible, and safe methodology to consistently obtain the correct values.

One method of testing criticality code validation techniques is to use a sample system with a known bias as a test application and determine if the methods employed can reproduce the know bias. In this summary, a low-enriched uranium (LEU) lattice critical experiment with a known bias is used as the test application, and numerous other LEU experiments are used as the benchmarks for the criticality code validation exercises using traditional and advanced parametric techniques. The parameters explored are enrichment energy of average lethargy causing fission (EALF), and the TSUNAMI integral index c_k with experiments with varying degrees of similarity."

TSUNAMI Sensitivity and Uncertainty Analysis Methods for Criticality Code Validation, Brad Rearden, Don Mueller, Steve Bowman, SCALE 5 Training Course, ORNL, October 30-November 2, 2006

"TSUNAMI - Tools for Sensitivity and Uncertainty Analysis Methodology Implementation"

13 List of Preparers

Alan Wells, PHD – Dr. Wells has over 35 years of experience in the nuclear industry specifically in the areas of criticality safety, shielding, structural, containment and thermal analysis for irradiated fuel shipping casks, irradiated fuel handling and rod consolidation equipment, and other nuclear equipment. Dr. Wells wrote a Full Burnup Credit Topical Report for the Electric Power Research Institute in 2010. Dr. Wells led the design of the NAC-LWT and NAC-STC transportation casks and the NAC-S/T storage cask that were licensed by the U.S. NRC, and he has had numerous contacts with the U.S. NRC/NMSS in regulatory matters. As Chief Engineer of Nuclear Assurance Corporation (NAC), Dr. Wells was responsible for the certification amendments for the NLI-1/2 and NAC-1 (NFS-4) truck casks. Dr. Wells coordinated the use of truck casks to ship spent research reactor fuel through the Panama Canal with the China AEC and Canal authorities as part of a U.S. DOE/DOS project. Dr. Wells is the author of the structural design and testing sections of the Cask Designer's Handbook (ORNL/M-5003) of the Transportation Technology division of ORNL. He is a co-author of the American Nuclear Society Standards Committee 8.21, which has developed a standard for criticality safety for fixed neutron absorbers. Dr. Wells is currently a member of the ANS 8.27 writing group for Burnup Credit as an EPRI representative. He also participated in a 2003 re-write of NRC Regulatory Guide 1536 for spent fuel storage casks and revised the Principal Design Criteria chapter. Dr. Wells is the author of the structural design and testing sections of the Cask Designer's Handbook (ORNL/M-5003) of the Transportation Technology division of ORNL.

Dominic Napolitano, PHD – Dr. Napolitano has 29 years of experience in the US nuclear industry, including 10 years in PWR reactor physics, 12 years in spent fuel transport and storage cask design, and 7 years in power plant radiological evaluations. His areas of expertise include: software V&V, shielding, radiological, criticality and reactor analyses as well as technical training in these areas. He is currently involved in burnup credit criticality evaluations of an LWR storage and transport cask for Equipos Nucleares SA and had validated burnup credit criticality methods under contracts with the Electric Power Research Institute and Sandia National Laboratories.