Log Number: 5.2MD12
Topic Area Number: 5.2
Oral Presentation

Computational Physics and Engineering Division

Investigation of Burnup Credit Issues in BWR Fuel

Mark D. DeHart and Bryan L. Broadhead
Oak Ridge National Laboratory,*
P. O. Box 2008,
Oak Ridge, TN 37831-6370

e-mail: dehartmd@ornl.gov
phone: (423) 576-3468
fax: (423) 576-3513

e-mail: broadheadbl@ornl.gov
phone: (423) 576-4476
fax: (423) 576-3513

Submitted to the
ICNC'99
Sixth International Conference on Nuclear Criticality Safety,
September 20–24, 1999,
Palais des congrès . Versailles . France

* Managed by Lockheed Martin Energy Research Corporation under contract DE-AC05-96OR22464 for the U.S. Department of Energy.

INVESTIGATION OF BURNUP CREDIT ISSUES IN BWR FUEL

Mark D. DeHart and Bryan L. Broadhead
Oak Ridge National Laboratory, USA

Abstract

Calculations for long-term-disposal criticality safety of spent nuclear fuel requires the application of burnup credit because of the large mass of fissile material that will be present in the repository. Burnup credit calculations are based on depletion calculations that provide a conservative estimate of spent fuel contents, followed by criticality calculations to assess the value of k_{eff} for a spent fuel cask or a fuel configuration under a variety of probabilistically derived events. In order to ensure that the depletion calculation is conservative, it is necessary to both qualify and quantify assumptions that can be made in depletion models used to characterize spent fuel. Most effort in the United States this decade has focused on burnup issues related to pressurized-water reactors. However, requirements for the permanent disposal of fuel from boiling-water reactors has necessitated development of methods for prediction of spent fuel contents for such fuels. Concomitant with such analyses, validation is also necessary. This paper provides a summary of initial efforts at the Oak Ridge National Laboratory to better understand and validate spent fuel analyses for boiling-water-reactor fuel.

Introduction

Although much effort has been dedicated to the study of the issues of burnup credit over the past decade, U.S. studies to-date have primarily focused on spent pressurized-water-reactor (PWR) fuel. The current licensing approach being taken by the U.S. Department of Energy for burnup credit in transportation seeks approval for PWR fuel only, because (1) the largest fraction of commercial spent fuel in the United States is PWR fuel, and (2) burnup credit is not as important for the transportation of boiling-water-reactor (BWR) spent fuel. However, attention has recently refocused on spent BWR fuel, due to international needs and criticality issues related to permanent disposal within the United States.

Burnup credit relies on the accurate prediction of spent fuel isotopic contents through depletion analyses. Validation is required to quantify biases and uncertainties between analytic predictions and measured isotopics. However, the design and operational aspects of BWRs result in a more heterogeneous and time-varying reactor configuration than those of PWRs. Therefore, BWR spent fuel analyses and validation efforts are significantly more complicated than those of their PWR counterparts. BWR spent fuel assemblies are manufactured with variable enrichments both radially and axially, are exposed to time-varying void distributions, contain burnable poison rods, and are subject to partial control blade insertion during operation. The latter is especially true in older fuel assemblies. Depletion methods used for characterization of spent fuel have typically been developed and validated for more homogeneous PWR fuel assemblies and must be reassessed for BWR configurations to determine a method to conservatively estimate the isotopic contents of spent fuel.

In support of the ongoing efforts of the Waste Package Design team at the Yucca Mountain Project in the United States [1], Oak Ridge National Laboratory (ORNL) has undertaken a study of the application of current spent fuel characterization methods to BWR analyses, based on the SCALE SAS2H sequence [2]. The purpose of this study is to develop recommendations for a methodology that will provide a conservative validated approach for BWR spent fuel depletion. The study will proceed in several phases in an attempt to refine the process as physical understanding is gained. This paper will discuss the findings of the first phases of the study and will describe plans for future work. Because of the substantial volume of data analyzed in this work, this paper is intended as a summary of the most significant findings to date, rather than a comprehensive study of BWR depletion. The work will be documented in detail in upcoming ORNL reports.

BWR Fuel Assembly Analyses

Depletion calculations for this study were performed based on a GE 8 × 8 assembly, as illustrated in Figure 1. Assembly data and operating history data applied in these analyses are based on neutronic and thermal-hydraulic data for a fuel assembly burned in Quad Cities Unit 2 during Cycles 10 through 14. Data were supplied to ORNL, modified to protect proprietary information [3], but nonetheless represent operational data for a real BWR assembly. For identification purposes, this assembly has been designated as Assembly ZZ.

Assembly ZZ, with an average enrichment of 3.2 wt % ^{235}U, contains 60 fuel elements with 10 different enrichments, including 9 rods containing 3 wt % Gd_2O_3. The various initial enrichments and locations of the fuel are indicated in the figure. This assembly design also contains a large central water hole, an outer casing ("can") and bypass moderator. Additionally, the assembly has a non-uniform axial loading, with natural U fuel reflectors comprising the last 15.24 cm of each end of the fuel. The non-uniform and asymmetric fuel loading of this assembly are expected to provide a severe test of the modeling capability of the SAS2H sequence of SCALE.

Calculations were performed to assess one-, two- and three-dimensional (1-, 2- and 3-D) effects. Detailed 2-D HELIOS [4] analyses were compared with 1-D cylindrical approximations of the assembly performed using SAS2H. SAS2H uses a 1-D transport solution (XSDRNPM) to generate 3-group fuel-averaged fluxes, which are used in the point depletion ORIGEN-S code. Studies focused on the effect of fuel homogenization and the treatment of burnable poison rods (BPRs) in the explicit 2-D and approximate 1-D models. Then, using the 1-D approach to calculate number densities for a set of axially varying burnups, 3-D KENO-V.a criticality calculations were performed to assess the effect of various axial zoning schemes and depletion assumptions on the calculated value of k_{inf} assuming an infinite lattice of fuel assemblies. Results of both sets of calculations are described in the following sections.

1-D vs. 2-D: The Effect of Homogenization

To test the ability of SAS2H to model such an inhomogenous BWR fuel assembly, depletion calculations were performed using both SAS2H and the HELIOS computer code package. Comparisons between these two codes, in tandem with variations in the HELIOS assembly model, were performed to assess the effect of assembly heterogeneity. HELIOS is a widely used tool for reactor fuel management analysis and has been validated for a number of reactor types, including many BWR fuel designs. Even though SAS2H is limited to simple, 1-D transport analysis assuming a single fuel type, HELIOS can perform pin-by-pin depletion calculations based on a 2-D transport solution. Although a code-to-code comparison lacks the quantification of a direct comparison to measured spent fuel data, such a comparison allows a study of the relative behavior of the two codes, and can provide an understanding of the effect and magnitude of the assumptions required for SAS2H analysis of such a system.

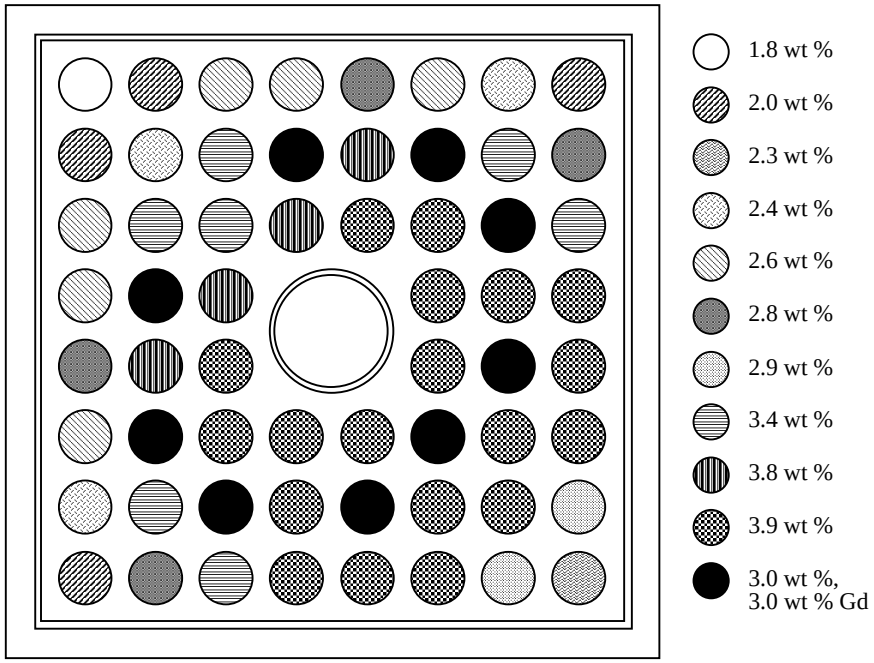

Figure 1. GE 8 × 8 assembly design used in this work.

The SAS2H model for a fuel assembly is limited to a 1-D radial model with a single smeared fuel region (see Fig. 2). Because it is impossible to represent the 9 distributed gadolinium-bearing fuel rods (Gd rods), the SAS2H model assumes a single Gd rod and clad, surrounded by smeared fuel, that represents one-ninth of the assembly fuel volume, bounded by corresponding volumes of assembly container and bypass moderator materials. Because it is not possible to model both the Gd rod and the central water hole, the volume of the water hole is included in the volume of the bypass moderator. The smeared fuel mixture includes fuel (3.2 wt % ^{235}U), clad, and moderator. Calculations are performed to obtain cell-weighted cross sections for the corresponding pin-cell model. Hence, only a single fuel enrichment is possible.

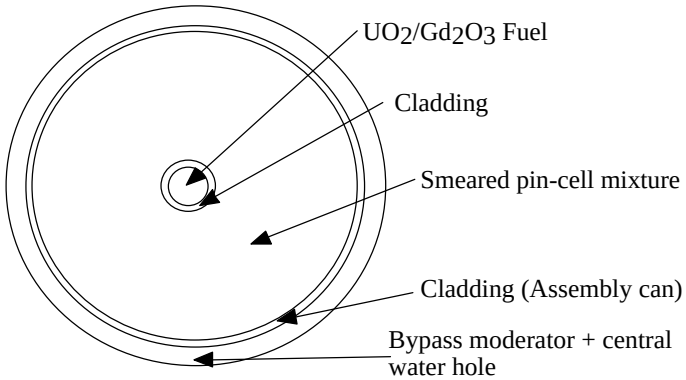

Figure 2. SAS2H model for 8 × 8 BWR fuel assembly (not drawn to scale).

Because HELIOS is able to model geometric effects exactly, the HELIOS model of the BWR assembly is exactly as shown in Fig. 1. No geometric or fuel enrichment approximations are necessary.

Calculations were performed using typical operational parameters for temperatures, and with a moderator density of 0.74 g/cc. Continuous operation at a power level of 30.9 MW/MTU to a total burnup

of 40 GWd/MTU was assumed, followed by a 5-y cooling time. Results for selected nuclides are given in Table 1 for assembly-averaged isotopics calculated by both HELIOS and SAS2H, along with the percentage difference between the two (SAS2H relative to HELIOS). These results show that even with the considerable approximations made in the SAS2H model, the code is for the most part within 10% of HELIOS predictions for important actinides and fission products. In general, SAS2H is overpredicting nuclide concentrations relative to HELIOS, with the significant exception of ^{235}U. The fact that ^{235}U is overburned and ^{238}U is underburned in the SAS2H calculation indicates a softer spectrum in the SAS2H model. However, this is contradicted by the higher Pu concentrations predicted by the SAS2H model. The exact reason for this behavior is unclear.

Table 1. Nuclide densities (in gram-atoms) calculated for the 8 × 8 BWR assembly.

Nuclide	HELIOS	SAS2H	% Diff.	Nuclide	HELIOS	SAS2H	% Diff.
U-234	5.99E-01	6.46E-01	7.85%	Tc-99	9.36E+00	9.79E+00	4.59%
U-235	1.60E+01	1.56E+01	-2.50%	Ru-101	9.10E+00	9.22E+00	1.32%
U-236	1.89E+01	1.92E+01	1.59%	Rh-103	5.01E+00	5.17E+00	3.19%
U-238	3.91E+03	3.97E+03	1.53%	Ag-109	8.63E-01	1.00E+00	15.87%
Pu-238	5.81E-01	6.62E-01	13.94%	Sm-147	1.67E+00	1.66E+00	-0.60%
Pu-239	1.42E+01	1.50E+01	5.63%	Sm-149	1.64E-02	1.74E-02	6.10%
Pu-240	1.01E+01	1.03E+01	1.98%	Sm-150	2.15E+00	2.29E+00	6.51%
Pu-241	3.01E+00	3.17E+00	5.32%	Sm-151	5.36E-02	6.42E-02	19.78%
Pu-242	2.77E+00	3.17E+00	14.44%	Sm-152	1.08E+00	1.09E+00	0.93%
Am-241	9.27E-01	9.83E-01	6.04%	Nd-143	5.21E+00	5.34E+00	2.50%
Am-242m	1.55E-03	1.91E-03	23.23%	Nd-145	5.36E+00	5.49E+00	2.43%
Am-243	4.72E-01	6.00E-01	27.12%	Eu-151	2.14E-03	2.57E-03	20.09%
Np-237	1.45E+00	1.76E+00	21.38%	Eu-153	8.61E-01	9.39E-01	9.06%
Mo-95	9.36E+00	9.61E+00	2.67%	Gd-155	2.21E-02	2.40E-02	8.60%

Additional calculations are proceeding to assess the performance of SAS2H relative to HELIOS with different moderator densities and different modeling approximations. Also, calculations are being performed to assess the effect of the presence of the Gd rod in the SAS2H model. Calculations to date indicate that SAS2H has a larger error relative to HELIOS early in an exposure history due to inadequacies in Gd depletion in the SAS2H approach. However, these errors are ameliorated with increasing burnup after the poison rod has been depleted.

SAS2H has been compared to HELIOS in earlier validation work for UO_2 fuel samples obtained from a MOX assembly design [5]. It is worth noting that the differences between actinides in the earlier work are consistent with those shown in Table 1. Additionally, for several actinides (^{236}U, ^{238}Pu, ^{240}Pu, and ^{237}Np), SAS2H was in better agreement with experimental measurement than HELIOS. Thus, code-to-code differences shown in Table 1 do not necessarily indicate limitations in the SAS2H approach for BWR spent fuel characterization.

Based on the above results and other ongoing work, the initial indication is that the approximations in the 1-D SAS2H model provide an adequate representation of depletion dynamics for a heterogeneous 2-D BWR assembly. Although not as rigorous as an explicit model, the 1-D approximation appears to yield consistent results such that a reasonable bias and uncertainty could be determined in the estimation of assembly-averaged isotopic concentrations. The simplicity and relative speed of the SAS2H approach for modeling complicated systems is preferred over more approaches. The following section describes calculations performed based on SAS2H depletions to assess the sensitivity of k_{inf} to depletion modeling assumptions.

Sensitivity Calculations in Three Dimensions

The goal in this work was to determine the sensitivity of k_{inf} for an infinite lattice of fuel assemblies to various depletion assumptions. This goal is accomplished by using a series of 1-D SAS2H depletion calculations to represent axial burnup at each of several axial regions based on the local burnup. These isotopic concentrations are then used to make an approximate 3-D model of an axially burned fuel, for which k_{inf} can be calculated. This approach lets one determine which depletion assumptions are conservative in estimating criticality (i.e., the depletion model that results in the highest predicted value of k_{inf} for spent BWR fuel). This process was performed at ORNL in earlier analysis of PWR spent fuel [6]. The earlier work served as a starting point for much of the work described here.

The Axial Burnup Model

It has been well established that the normal variation of burnup along the length of a spent fuel assembly results in a shift of the peak fission density away from the center, where the fuel is most reactive at the beginning of life. However, as the most reactive (less depleted) region of the fuel shifts away from the center, axial leakage also increases. Thus, in order to assess the net reactivity worth of spent fuel, it is necessary to develop a 3-D representation of the assembly in which a burnup profile is included. To approximate the continuous distribution of burnup in a burned fuel assembly, a model was developed in which a single fuel assembly was divided into multiple axial regions. A burnup profile was assumed for the fuel assembly and approximated as discrete burnup intervals, representing the average burnup across the length of each interval. Depletion calculations were performed for each interval, assuming identical operating histories but with varying specific power such that the correct burnup was obtained in each region at the end of the depletion calculations. Baseline operating parameters were obtained from detailed Assembly ZZ data.

Prior to initiating sensitivity calculations, calculations were performed to identify an axial model with the minimum number of axial regions to accurately capture axial effects. This step was done to reduce the number of calculations required to perform sensitivity calculations, since a separate depletion calculation is necessary for each axial zone in a 3-D model. The fact that burnup is fairly constant near the center of BWR fuel suggests that a number of central axial zones can be combined with no significant effect on the axial model — demonstrated for PWR fuel in Ref. 6. The initial model was based on 24 axial zones (the form in which Assembly ZZ data were provided); one axial zone at each end for the low-enrichment blanket fuel, and 22 zones of standard fuel. Depletion calculations were performed using operational parameters, the assembly cross-plane approximated using the average fuel enrichment as described earlier. The predicted isotopics for each axial zone were then fed into a corresponding 24-zone KENO-V.a model to calculate the value of k_{inf} for the assembly. Next, the two central zones were collapsed and properties (specific power, temperatures, etc.) were averaged to create a single central zone. SAS2H depletion calculations and the KENO-V.a criticality calculation were repeated in a similar manner. This collapsing procedure was repeated by successively increasing the number of axial zones combined in the center region, to a minimum 3-zone model, required to maintain the properties of the blanket material at the two ends. The entire process was performed for burnups of 40 and 60 GWd/MTU. In both sets of analyses, it was demonstrated that the calculated value of k_{inf} for the assembly was not sensitive to combined cells for models containing 11 or more axial zones. Thus, remaining sensitivity calculations were performed using an 11-zone axial representation, consisting of the original five axial zones at either end, with the central 14 zones combined into a single axial zone with averaged properties.

Moderator Density

Reference 6 demonstrated a strong coupling between depletion predictions and the assumed moderator density. The nature of moderator density variations in BWR fuel designs makes this a more significant issue. Assembly ZZ data provide moderator density as a function of burnup for specific statepoints over the entire life of a fuel assembly. These data show that burnup profiles remain relatively constant over time, although for this assembly the moderator density in the upper regions initially decreased, then increased to a maximum near the end of life. Such variations are expected to be operationally dependent. Thus, for simplicity it would be best to assume a bounding value for a moderator density over the life of the fuel assembly. This study was performed to assess the *direction* of conservatism (i.e., whether the lowest or highest estimate represents the conservative bound), and to determine the magnitude of this conservatism as a function of burnup. This goal was accomplished by calculation assuming nominal conditions, but with a fixed uni form moderator density varying from 0.2 to 0.8 g/cm^3. SAS2H calculations were performed assuming initial enrichments of 3.0, 3.75, and 4.5 wt % ^{235}U, and for burnups of 10, 30, and 50 GWd/MTU. Results are plotted in Fig. 3, with k_{inf} normalized to the value computed for minimum density for the purposes of comparison. The results show a close-to-linear response; non-linear variations are felt to be due to stochastic effects in the Monte Carlo k_{inf} calculation; hence linear fits were applied to the data to more easily illustrate trends. No error bars are plotted because the uncertainty in each k_{inf} calculation is on the order of the size of the plot symbols. These results clearly indicate that k_{inf} is maximized (the depletion model becomes more conservative) with the lowest moderator density, and that conservatism increases with increasing fissile depletion. It is believed that this is a spectrum-driven effect, resulting from increased Pu production and fission, and a concurrent reduction of ^{235}U depletion due to spectral hardening that occurs as moderator density is decreased.

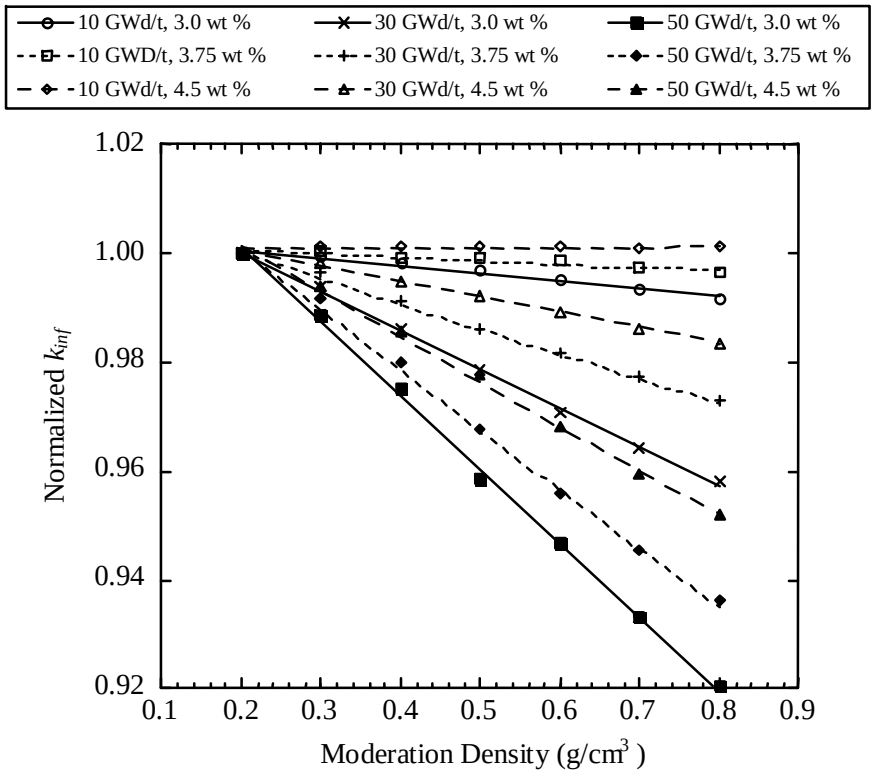

Figure 3. Effect of depletion moderator density on k_{inf} .

Fuel Temperature

SAS2H calculations were performed to assess the effect of assumed fuel temperature in much the same manner. Data from Assembly ZZ operations were assumed for depletion calculations, but constant fuel temperatures were assumed in each depletion model. Sets of calculations were performed, varying the fuel temperature from 500 to 1300 K for various burnup states and initial enrichments. Results are shown in Fig. 4, with k_{inf} normalized by the value calculated for 500 K for each burnup/enrichment pair. Again, the response appears to be linear, as illustrated by the linear fits to the data. Error bars due to Monte Carlo uncertainty are on the order of the size of each of the plot symbols. The results demonstrate that the estimate of k_{inf} is increased with an increase in the assumed fuel temperature during depletion, and that, like moderator density effects, the effect of temperature increases with fissile depletion. This behavior is probably due to doppler broadening in ^{238}U, resulting in enhanced Pu production and less ^{235}U depletion, similar to spectral hardening effects in decreased moderator densities. The magnitude of the reactivity effect of temperature is relatively small, however, especially given that the range of fuel temperatures in operating reactors is much smaller than the range studied here. The effect is also dwarfed by the magnitude of moderator density effects.

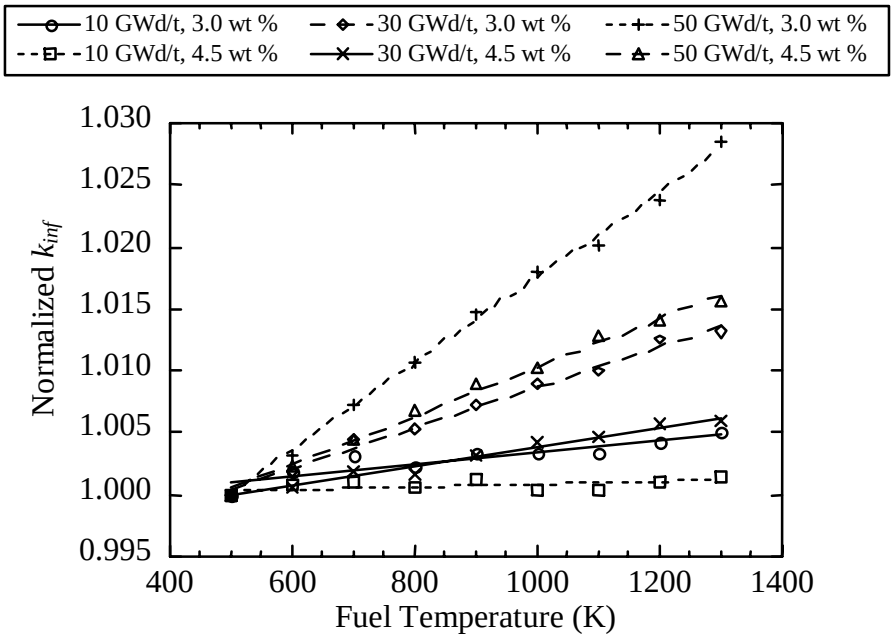

Figure 4. Effect of fuel temperature applied during depletion calculations.

Operating History

The final issue examined in this study was the effect of the reactor operating history. Since the operating history of spent fuel varies significantly between different assemblies, it is necessary to determine an operating history that is bounding in terms of its effect on criticality, or to define a simple operating history and have a quantified margin associated with that history that will bound the effect of operational variations.

The calculations performed here were not an attempt to define a limiting profile or an appropriate margin; rather, they were performed to determine which operational parameters (e.g., specific power variations over time, downtime lengths, etc.) had the most significant effect on the reactivity worth of fuel

after a 5-y cooling period. Table 2 summarizes the operating histories studied. These histories essentially represent three burn cycles, with downtime assumed in the middle of each cycle in addition to inter-cycle downtime. Histories 1 through 11 were applied in PWR studies described in Ref. 6. History 12 was added to represent another operating pattern observed in BWR fuel, based on available Assembly ZZ operating data. History 13 represents the actual Assembly ZZ burnup history.

Table 2. Description of Operating Histories Analyzed.

1) Six 180-day full power periods, no downtime	2) Six 180-day full-power periods separated by 20-day down periods (10% downtime)
3) Six 180-day full-power periods separated by 45-day down periods (20% downtime)	4) Six 180-day full-power periods separated by 77-day down periods (30% downtime)
5) Six 180-day full-power periods, 10% downtime, 30% downtime in 3rd and 4th periods	6) Six 180-day full-power periods, 10% downtime, 30% downtime in 5th and 6th periods
7) Six 180-day full-power periods, 10% downtime, 720 day downtime between 3rd and 4th periods	8) Six 180-day full-power periods, 10% downtime, 720 day downtime between 5th and 6th periods
9) Six 180-day periods, 120% power in first two periods, 90% power in remaining periods, 10% downtime	10) Six 180-day periods, 120% power in 3rd and 4th periods, 90% power in remaining periods, 10% downtime
11) Six 180-day periods, 120% power in last two periods, 90% power in remaining periods, 10% downtime	12) Six 180-day periods, 120% power in first four periods, 60% power in remaining two periods, 10% downtime
13) Actual operating history profile from Assembly ZZ data	

As with the earlier calculations, operating history effects were calculated for a variety of initial enrichments and burnups, representing a range of fissile depletion. The full operating history was applied for each burnup state, by scaling the average specific power for each fuel depletion model. Results are plotted in Fig. 5. Behavior for histories 1 through 11 are consistent with trends observed in Ref. 6 and in a similar study that used 3-D criticality calculations [7], both for PWR analyses. However, it is clear that histories 12 and 13, with very low-power operation for the last fuel cycle, represent the most conservation depletion histories with respect to k_{inf} calculations.

It was postulated in Ref. 6 that the low-power operation results in a lower equilibrium concentration of ^{155}Eu during the last cycle, and that equilibrium conditions are reached during the length of a cycle. The lower ^{155}Eu concentrations at discharge result in lower ^{155}Gd inventories post-shutdown and a higher net fuel reactivity. These results presented here tend to confirm this hypothesis. The effect is likely compounded by other decay products that are daughters of important fission products (e.g., ^{147}Sm, which is produced by the decay of ^{147}Pm with a 2.6-y half-life). However, the net effect, for very highly burned fuel, is on the order of 0.5%.

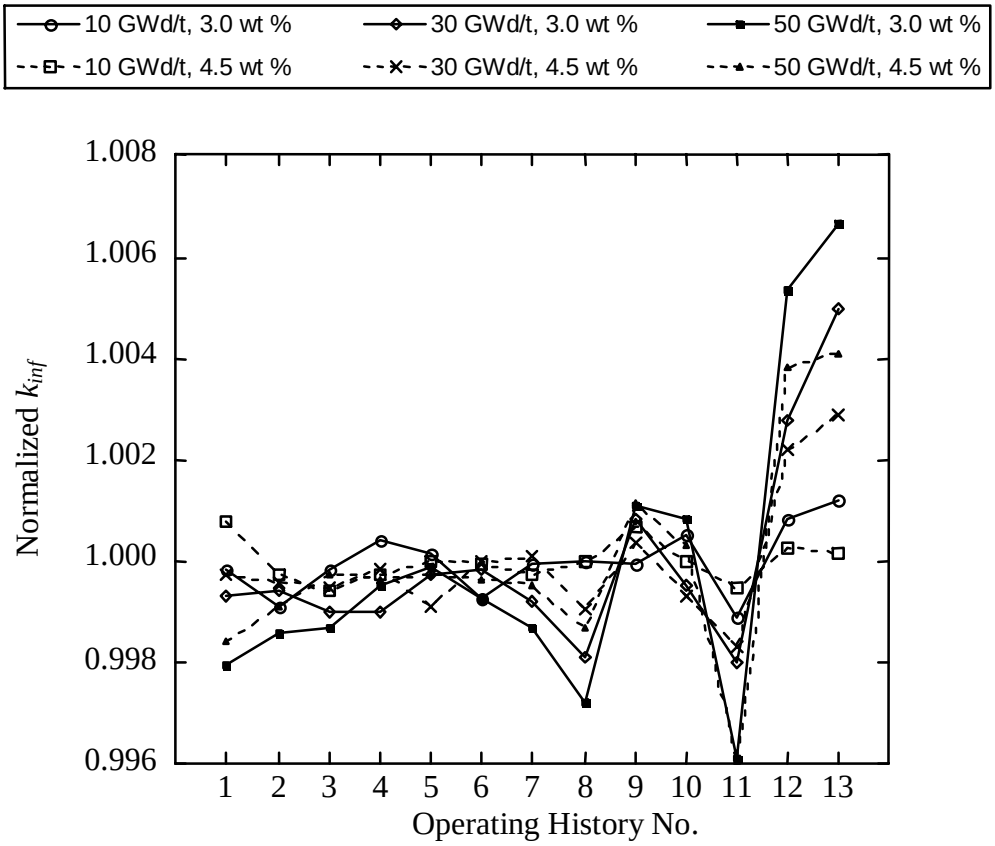

Figure 5. Effect of operating history assumed during depletion calculations.

Conclusions

This paper has described initial scoping analyses performed at ORNL to achieve a better understanding of the modelling requirements for BWR depletion calculations. Results completed at ORNL have been summarized here, and additional calculations are ongoing. Calculations completed to-date have demonstrated similar trends to those observed for PWR spent fuel calculations [6].

The effect of the heterogenous lattice of a BWR design does introduce errors in 1-D depletion analyses that were not a problem in the more homogeneous lattices common in PWR designs. Further work is needed to quantify the differences. Validation analyses performed for BWR fuel assemblies relative to chemical assay measurements have shown similar uncertainties, although much of the uncertainty is believed to be due to uncertainty in moderator densities in the vicinity of fuel samples [5, 8]. Additional comparisons between SAS2H and HELIOS are being performed under a variety of different conditions (e.g., varying moderator densities and control blade insertions) to assess the effect of those conditions on the accuracy of SAS2H relative to HELIOS.

Basic depletion parameters were examined in this study: moderator density, fuel temperature, and operating history. Trends observed here are consistent with those observed in the study of PWR depletion modelling, as is the magnitude of the effect, and there is believed to be no reason to expect significant differences in the approach required to conservatively model the depletion process between BWR and PWR designs.

The work reported herein has been based on code-to-code comparisons and parametric analyses of depletion models. However, such comparisons are of little value without direct comparisons to measured data. As mentioned earlier, Ref. 7 describes a set of calculations used to validate SAS2H against radiochemical assay data. However, those data are insufficient in depth or applicability for the general population of BWR fuel. Other candidate fuel measurements have been identified by ORNL staff and will hopefully be analyzed in the future. However, there remains a lack of well-qualified measurement data applicable to most modern BWR designs.

References

1. Yucca Mountain Site Characterization Project: Disposal Criticality Analysis Methodology Topical Report, MP/TR-004Q, Revision 0, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Las Vegas, Nevada, November 1998.

2. O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," Vol. I, Sect. S2 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 5 (ORNL/NUREG/CSD-2R5), March 1997. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

3. Personal communication, David Henderson, Waste Package Design, Framatome Cogema Fuel Company, Las Vegas, Nevada, 1999.

4. *HELIOS Documentation*, The Scandpower Fuel Management, December 1995.

5. B. D. Murphy, *Prediction of the Isotopic Composition of UO_2 Fuel from a BWR: Analysis of the DU1 Sample from the Dodewaard Reactor*, ORNL/TM-13687, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, October 1998.

6. M. D. DeHart, *Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages*, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996.

7. M. D. DeHart, *Parametric Analysis of PWR Spent Fuel Depletion Parameters for Long-Term Disposal Criticality Safety*, ORNL/TM-1999/99, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, in preparation.

8. O. W. Hermann and M. D. DeHart, *Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel*, ORNL/TM-13315, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, September 1998.