Overview of the Recent BWR Burnup Credit Project at Oak Ridge National Laboratory[1]

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ABSTRACT

Oak Ridge National Laboratory (ORNL) and the US Nuclear Regulatory Commission (NRC) have completed a five-year program to investigate burnup credit (BUC) for boiling-water reactor (BWR) spent nuclear fuel (SNF) stored in storage and transportation systems. The project examined both peak reactivity BUC, also sometimes called gadolinium credit*, and extended BWR BUC. Here* extended *refers to credit for burnups beyond that associated with peak reactivity. The findings related to peak reactivity BUC are summarized in NUREG/CR-7194, and four additional NUREG/CR documents present the results of the studies of extended BWR BUC. The first of these documents, NUREG/CR-7224, presents results of investigations to determine the effects of axial moderator density profiles, control blade use, and axial burnup profiles. Studies on the impact of core operating conditions and assembly-specific depletion conditions are addressed in NUREG/CR-7240, validation of depleted SNF isotopic predictions is addressed in NUREG/CR-7251, and validation of k_{eff} calculations for extended BWR BUC are discussed in NUREG/CR-7252. A summary of the entire project, including major conclusions regarding each of the studies, is included in this paper.*

KEY WORDS

Burnup credit, boiling water reactor

1. INTRODUCTION

Applicants for certificates of compliance for spent nuclear fuel (SNF) transportation and dry storage systems, referred to as *casks* in this paper, perform analyses to demonstrate that these systems are adequately subcritical per the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Parts 71 and 72 [1]. These analyses may credit the reduction in assembly reactivity caused by depletion of fissile nuclides and buildup of neutron-absorbing nuclides during power operation. This credit for reactivity reduction during depletion is commonly referred to as *burnup credit* (BUC). US Nuclear Regulatory Commission (NRC) staff perform reviews of BUC for pressurized-water reactors according to the guidance in the Division of Spent Fuel Storage and Transportation, *Interim Staff Guidance-8*, *Revision 3, Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks* [2].

BUC for boiling water reactor (BWR) SNF is not addressed in ISG-8, but research was performed at Oak Ridge National Laboratory (ORNL) between 2013 and 2018 to investigate the principal phenomena relevant

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to regulation of BWR BUC. This research project was organized based on the recommendations of NUREG/CR-7158 [3]. The project has consisted of 10 tasks divided into two phases. Phase I contained Tasks 1 and 2, for the study of BWR peak reactivity analysis for transportation and storage casks. Phase II consisted of Tasks 3–10, which investigated extended BWR BUC for burnups beyond peak reactivity. A brief summary of each of the first nine tasks is presented in Table I, which provides a list of the 10 tasks, their subjects, and the final report documenting the results of the studies.

All calculations were performed with SCALE, although different versions were used for different tasks. SCALE 6.1 [4] was used for Tasks 1–5, SCALE 6.2.1 [5] was used for Tasks 6 and 7, and SCALE 6.2.2 [6] was used for Tasks 8 and 9. Each task used the most current release available at the time the work was performed. Depletion calculations were primarily performed with the TRITON (t-depl) sequence, although some tasks used the STARBUCS sequence instead. The isotopic validation calculations in Task 8 were performed with Polaris instead of TRITON. All cask effective neutron multiplication factor (k_{eff}) calculations were performed using KENO V.a to model the GBC-68 computational benchmark cask [7].

The subsequent sections of this paper provide summaries of the important conclusions of each of the nine technical tasks in the program. These summaries are generally taken directly from the referenced reports. This paper is primarily a collection of all the conclusions and recommendations generated throughout this project.

Table I. Summary of the 10 tasks in the BWR BUC project

Phase	Task	Description	Final report
I	1	Evaluate use of peak reactivity analysis in transportation and storage systems	NUREG/CR-7194 [8]
	2	Evaluate validation of peak reactivity and burned fuel composition calculations	
II	3	Recommendations for treatment of axial moderator density distributions	NUREG/CR-7224 [9]
	4	Recommendations for treatment of control blade use during depletion	
	5	Recommendations for identification and use of axial burnup distributions	
	6	Recommendations for use of reactor operating parameters in depletion	NUREG/CR-7240 [10]
	7	Study for guidance and handling of operating parameter correlations affecting fuel depletion	
	8	Recommendations for validation of BWR burned fuel isotopic composition calculations	NUREG/CR-7251 [11]
	9	Recommendations for validation of BWR burned fuel k_{eff} calculations	NUREG/CR-7252 [12]
	10	Document BWR BUC guidance	Letter report to NRC

2. TASK 1: EVALUATE USE OF PEAK REACTIVITY ANALYSIS IN STORAGE AND TRANSPORTATION CASKS

The purpose of this task was to examine the impacts of important BWR design and operating parameters on reactivity for burnups up to the peak reactivity. Sensitivity calculations were performed over a range of parameters, including gadolinium loadings and assembly patterns, void fraction, control blade insertion, fuel temperature, specific power, and operating history. All depletion calculations performed in these analyses are two-dimensional planar calculations, consistent with typical use of peak reactivity methods. This section summarizes the sensitivity studies results, lists the recommended analyses that should be presented as part of a criticality safety evaluation, and defines areas of future study. It is expected that the results presented here are applicable to all modern BWR fuel assembly design types.

a. Summary of Sensitivity Studies

A range of parameters was studied to identify which are important for BWR peak reactivity and to quantify the potential sensitivity of calculated cask reactivity to these parameters. The fuel assembly model used in this study was the GE14 fuel assembly, which has a 10 × 10 array of fuel pins and contains two large central water rods, each of which displaces four fuel rods. Full and vanished lattices with no gadolinium-poisoned pins were used as the base case for actinide-only (AO) BUC sensitivity studies. For the actinide and fission product (AFP) BUC sensitivity studies, six gadolinium-poisoned fuel pins with 2.0 wt% gadolinium oxide were used as the base case. The results of the sensitivity studies are summarized below and shown in Table II and Table III for AO and AFP, respectively.

Table II. Summary of parameters affecting discharged reactivity for BWR peak reactivity analysis considering the AO nuclide set

Parameter	Range Studied	Reactivity Impact (%Δ*k*)
Fuel composition modeling	AEAI[a], PEAI[b], PEPI[c]	0.3 (AEAI)
Number of gadolinium pins	0 to 8 pins (0 pins base case)	~0 (≤0.06 at 7 GWd/MTU)
Loading of gadolinium pins	0 to 10 wt%	0 to 0.5 (9 to 25 GWd/MTU)
Gadolinium pattern	15 patterns	~0 (±0.03 at 7 GWd/MTU)
Moderator void fraction, unrodded	0 to 80% (40% base case)	-0.4 to +0.7 (0% void to 80% void)
Moderator void fraction, rodded	0 to 80% (40% base case)	-0.6 to +1.1 (0% void to 80% void)
Control blade insertion	unrodded vs. rodded (full depletion)	≤ 1 (rodded)
Fuel temperature	850 to 1100 K (950 K base case)	~0.0020%Δ*k* / 100 K
Specific power	25 to 45 W/g (25 W/g base case)	0.1 (35 and 45 W/g)
Operating history	Four histories (FP base case)	-0.1
Cask model dimension	2D, 3D (2D base case)	-0.2

[a] lattice-average enrichment (AE) and average isotopics (AI)
[b] pin-wise enrichment (PE) and average isotopics (AI)
[c] pin-wise enrichment (PE) and pin-wise isotopics (PI)

Table III. Summary of parameters affecting discharged reactivity for BWR peak reactivity analysis considering the AFP nuclide set

Parameter	Range studied	Reactivity impact (%Δk)
Fuel composition modeling	AEAI, PEAI, PEPI	0.5 (AEAI)
Number of gadolinium pins	2 to 8 pins (6 pins base case)	+1.3 to -0.2 (2 to 8 pins)
Loading of gadolinium pins	2 to 10 wt% (2 wt% base case)	-3 to -11 (4 wt% to 10 wt%)
Gadolinium pattern	15 patterns	±0.25 at peak reactivity (~7.5 GWd/MTU)
Moderator void fraction, unrodded	0 to 80% (40% base case)	+0.2 to -0.2 (0% void to 80% void)
Moderator void fraction, rodded	0 to 80% (40% base case)	+0.1 to -0.2 (0% void to 80% void)
Control blade insertion	unrodded vs. rodded (full depletion)	~1.0 (rodded)
Fuel temperature	850 to 1100 K (950 K base case)	0
Specific power	25 to 45 W/g (25 W/g base case)	-0.1 (peak reactivity)
Operating history	four histories (FP base case)	0.1 (HP case)
Cask model dimension	2D, 3D (2D base case)	-0.2

b. Recommended Minimum Analysis

A range of parameters was investigated in Task 1, and a set of conclusions was developed related to which parameters should be examined for each licensing analysis. The primary parameter of interest in this category is the gadolinium pattern used in the analysis and the way this pattern interacts with control blade insertion. It is conceivable that some gadolinium patterns will be more reactive than others. More importantly, the power distribution shifts caused by control blade insertion drive the changes in residual gadolinium, and hence discharged fuel reactivity. These effects will be sensitive to the gadolinium pattern. Certain other key parameters do not need to be studied further to identify the direction of conservatism, but justification that the values used are appropriate should be provided as part of an application. The parameters that do not need further investigation for the GE14 design considered in this work include initial or depleted composition modeling, void fraction, fuel temperature, and operating history.

3. TASK 2: VALIDATION OF PEAK REACTIVITY ANALYSES

The validation of both k_{eff} calculations and depleted isotopic fuel composition determinations was considered in Task 2. The results of each of these studies are presented herein.

c. Summary of Validation of Cask Reactivity Calculations

The original assessment of k_{eff} calculation validation was performed with SCALE 6.1 [4] in [8]. An updated assessment was performed with SCALE 6.2.2 [6] in [12] to capture the impact of updated covariance data on the validation assessment. The results of both studies are included here. The recommendations for BWR peak reactivity validation are analogous to those provided in NUREG/CR-7109 [13] for PWR validations.

Sample validations were presented in NUREG/CR-7194 [8] and NUREG/CR-7252 [12] using a nontrending method, a traditional trending method for enrichment and the energy of the average lethargy of neutrons causing fission (EALF), and a c_k trending method. The resulting bias and bias uncertainty values were similar for each of the methods, although some deviations appear with the larger experiment set used in [12]. Although the number of critical experiments with satisfactory similarity is sufficient for statistical analysis, the lack of several important nuclides in the validation suite is problematic. Reactivity margins (penalty factors) should be applied to extend the area of applicability to include these nuclides. A sensitivity/uncertainty analysis approach similar to that described in NUREG/CR-7109 [13] was used to develop these factors in NUREG/CR-7194 [8] and NUREG/CR-7252 [12]. The calculations show that these penalty factors are fairly small and vary somewhat with different covariance libraries. It is unlikely that additional conservatism in them would have a noticeable impact on the overall conservatism of the analysis.

Validation calculations can be performed in compliance with the consensus ANSI/ANS standard on validation [14] and with prior NUREG/CR recommendations [15,16]. It was demonstrated that reactivity margins can be established that extend the validation area of applicability to account for some nuclides that cannot be validated with critical experiments at this time. Overall, validation of cask reactivity calculations, including fuel at peak reactivity, does not present a technical barrier that would prevent implementation of peak reactivity methods for demonstrating the criticality safety of BWR SNF in storage and transportation casks.

d. Validation of Spent Fuel Nuclide Compositions

A method for determining the uncertainty in k_{eff} calculations associated with the BWR SNF nuclide concentrations near peak reactivity was demonstrated for the GBC-68 cask model considering the vanished lattice configuration (see Section 5.1 of NUREG/CR-7194 [8] for details). This method is based on the direct application of measured BWR SNF composition data to estimate the uncertainties associated with the calculated nuclide composition used in BUC application models.

Publicly available experimental nuclide data for BWR fuel near peak reactivity currently limits BUC analyses to consider AO fuel compositions. Present radiochemical assay data lack sufficient fission product (FP) data to enable validation for the AFP isotope set. Similarly, applicable data for residual ^{155}Gd are limited, and only three samples obtained from PWR fuel samples were evaluated in this work. However, the large variability in validation results for ^{155}Gd and the statistical penalty associated with the low number of measurements effectively eliminate any potential credit that can be obtained based on existing measurement data.

The estimated margin for k_{eff} uncertainty in the calculated actinide compositions for the AO cask configuration is 0.0165 Δk with a 95% confidence level. The actinide compositions exclude ^{241}Am due to large measurement uncertainties. Addition of FP measurement data and improved ^{155}Gd data in the range of peak reactivity could support credit for a substantial fraction of the reactivity worth from the omitted BUC nuclides, which represent approximately 3.64% of the total reactivity worth.

4. TASK 3: GUIDANCE FOR TREATMENT OF AXIAL MODERATOR DENSITY DISTRIBUTIONS

This section summarizes studies examining the effects of axial moderator density profiles on extended BWR BUC [9]. Two separate studies were performed to examine the impact of axial moderator distributions: a temporal fidelity study and an axial moderator density study. The temporal fidelity study was required to examine the modeling approaches necessary for capturing the reactivity impacts of temporal variations in the axial moderator density profile, and the axial moderator density study was performed to determine the effect of these axial profiles once they were established. The results of the temporal fidelity study, presented in Section 4.5.1 of NUREG/CR-7224 [9], indicate that cycle average moderator density profiles can be used in depletion calculations with the addition of a modest reactivity penalty. The axial moderator density profile study results, presented in Section 4.5.2 of NUREG/CR-7194 [9], show that low moderator densities in the top few nodes of the profile lead to conservative reactivity determinations. Averaging the moderator densities across assemblies or nodes is not appropriate and will lead to nonconservative reactivity determinations.

Following are recommendations from these studies.

- A cycle-averaged moderator density can be used in each node of an axial moderator density profile for depletion calculations for simplicity, with an appropriate penalty for conservatism. A reactivity penalty of 0.25% Δk_{eff} is recommended to cover potential differences between detailed and cycle average moderator density treatments in depletion calculations. This penalty is nearly twice the magnitude of the largest discrepancy between detailed and average moderator density modeling observed in this study.
- A limiting axial moderator density profile will have low moderator densities in the top nodes of the assembly. Each applicant should present and defend the method used to identify limiting profiles. A limiting axial moderator density profile can be constructed by selecting the minimum density in each axial node from a collection of applicable actual profiles.
- Use of average moderator densities determined from consideration of multiple assemblies or multiple axial nodes will result in reactivity underprediction.
- A single moderator density value can be used conservatively in all nodes if it bounds the moderator densities in all nodes of the assembly.

5. TASK 4: GUIDANCE FOR TREATMENT OF CONTROL BLADE USE DURING DEPLETION

This section documents studies examining the effects of control blade insertion on extended BWR BUC [9]. Two sets of control blade histories were analyzed as part of this study: (1) hypothetical histories (HHs) to examine key effects of control blade insertion depth, length, and time in cycle, and (2) realistic histories (RHs) constructed from available operating data. The important effects ascertained using the HHs are compared with the results based on RHs to determine approximate differences associated with simulations of fuel assemblies over the entire irradiation. The results of the hypothetical control blade histories indicate three effects that result in increased cask reactivity: (1) control blades inserted deeply into the core, (2) control blades inserted near end of life, and (3) control blades inserted for longer periods of time (or burnup). In particular, cask reactivity is especially sensitive to control blades being inserted deeply into the core due to the axially top-peaked fission distribution (end effect) typical in BWR storage and transportation casks.

The results of this study indicate that neglecting control blade insertion during depletion could result in nonconservative cask criticality values of approximately 0.6% Δk. Additional operating data would be needed to generate a more generically applicable estimate of the cask reactivity impact of realistic control blade usage.

Key observations of this study are summarized as follows.

- Control blade insertions of 50% or less for the entire depletion have virtually no impact on cask reactivity. Deeper control blade insertions (80% of the active fuel length or higher) have a considerably greater impact than frequent shallower insertions.
- Although unrealistic, the most limiting case for the AFP isotope set is 92% blade insertion for the entire irradiation, which increases cask reactivity by 4.3% Δk compared with unrodded irradiation. The limiting case for the AO isotope set—full control blade insertion for the entire irradiation—is also unrealistic and results in a cask reactivity increase of 4.1% Δk compared with unrodded irradiation.
- Deep control blade insertions in the final third of life have greater impact (~1%) on reactivity than similar insertions earlier in life. Fuel assemblies are unlikely to experience significant operational periods with control blade insertion late in life because of lower reactivity and placement near the core periphery; therefore, the limiting cases examined here are considered conservative.
- Some fuel assemblies may experience exceptional control blade use as a result of suppressing assembly power because of leaking fuel rods or high assembly reactivity early in life.
- Based on the limiting realistic histories examined, a penalty of ~0.6–1.2% Δk_{eff} may be sufficient to account for control blade insertion effects. Applicants should provide a justification that any reactivity penalty taken is sufficient to account for rodded operations and should identify a process for handling assemblies that violate the underlying assumptions used in generating the penalty.

6. TASK 5: GUIDANCE FOR IDENTIFICATION AND USE OF AXIAL BURNUP DISTRIBUTIONS

This section documents studies performed to examine the effect of axial burnup distributions on extended BWR BUC [9]. The burnup range examined was 30–50 GWd/MTU. The lower burnup was selected to be higher than the burnups typical of peak reactivity for fuel lattices, whereas the upper bound is typical of discharge burnups for BWR assemblies. Detailed calculations were performed using 624 axial burnup profiles generated from a proprietary core follow data set. All 624 assemblies considered had natural enrichment blankets at the top and bottom of the assembly; most had 6-inch blankets at both ends of the assembly. A smaller subset of assemblies had 6-inch blankets on the bottom end of the assembly but 12-inch blankets at the top end. Analyses were performed both excluding and crediting the presence of the reduced enrichment blankets. The general trends in the results for both modeling approaches are similar, though significant margin is realized by taking credit for the blankets.

The profiles used in this study lead to a wide range of cask k_{eff} values. The most reactive profiles are generally the same with or without blankets for both isotope sets, as long as the profiles considered have the same axial features near the top end of the assembly. As an example of the effects that differences in axial features like blankets can have, the 12-inch blanket profiles are significantly more reactive than the 6-inch blanket profiles when no blanket is modeled.

The end effect was calculated for each profile at each burnup considered. A positive end effect indicates that the distributed burnup profile must be considered in the analysis, as it is more reactive than the uniform profile at the same burnup. Large positive end effects appear to be common for BWR fuel; all 624 profiles had positive end effects for the models neglecting blankets at all burnups, and at 50 GWd/MTU when blankets were modeled. The largest end effect for the profile with no blankets modeled ranges up to 12.7% Δk and the largest end effect considering blankets is 8.0% Δk. These end effects are larger than those typically seen in PWR BUC.

A method to identify potentially limiting burnup profiles was investigated in NUREG/CR-7224 [9**Error! Reference source not found.**]. It appears that for models with and without blankets, such an approach—based on summing the relative burnups of the top several nodes in a profile—can generate a reliable indication of the relative reactivity of profiles taken from assemblies with similar axial features near the top end of the assembly. For assemblies with no blankets modeled, it appears that the sum of relative burnups from a total of 18 inches near the top of the assembly is an optimum metric for ranking burnup profiles. The fuel length that must be considered is longer when blankets are modeled, probably on the order of 36 inches at the top end of the assembly. The method used by any applicant must be justified. It also appears that axial feature differences that are not near the top end—such as differences in the number and length of part-length rods or differences in the number and loading of gadolinium pins—do not need to be tracked to determine limiting axial burnup

profiles. The success of these basic methods in predicting relative reactivity can greatly simplify the work needed to determine the appropriate profile for use in a BWR BUC analysis.

The final aspect considered is the dependence of cask k_{eff} on the burnup of the assembly from which the profiles were taken. Three burnup bins were generated: 0–25 GWd/MTU, 25–40 GWd/MTU, and >40 GWd/MTU assembly average burnup at end of cycle (EOC). The separation of high-burnup assemblies into two bins at 40 GWd/MTU was largely arbitrary. For the profiles considered in this study, cask k_{eff} decreases as burnup increases. The use of burnup bins to ensure appropriate profiles in each burnup range is common in PWR BUC and will likely be useful in BWR BUC as well.

The following recommendations can be made based on the results of this study:

- The choice of axial burnup profiles can have significant impacts on cask reactivity. Applicants must provide justification for the profiles used in an analysis, including the burnups over which uniform and distributed burnup profiles are used.
- The limiting profile resulting from a set of available profiles is largely independent of the isotope set used. Axial blanket modeling approaches have only a small impact on identifying the limiting profile.
- Distributed burnup profiles must be considered for extended BWR BUC.
- The relative reactivity of different axial burnup profiles can be predicted reliably by considering the relative burnup in the top few nodes. Lower relative burnups lead to higher cask k_{eff} values. More nodes must be considered if axial blankets are modeled, and the relative ranking prediction is reliable only within a population of assemblies with blankets of the same length.
- Grouping axial burnup profiles into bins based on the EOC burnup of the assembly from which the profile was taken is likely to lower calculated cask k_{eff} values at higher burnups.

7. TASK 6: GUIDANCE FOR USE OF REACTOR OPERATING CONDITIONS

The calculated cask reactivity effects of reactor operating conditions are determined for a single assembly average burnup of 45.2 GWd/MTU [10]. The variations illustrated are expected to be similar for similar burnup values, but further analysis would be needed to apply these results to significantly different burnup values.

The directions and magnitudes of the impacts on cask reactivity are summarized here for each reactor operating parameter assessed in NUREG/CR-7240 [10]. As expected, the impacts of the reactor operating parameters are small relative to the impacts of control blade usage, axial coolant density, and axial burnup profile [9].

a. <u>Fuel Temperature</u>

- Cask reactivity increases linearly with increasing fuel temperature over the range of 596–1296 K for each of the AO and AFP isotope sets.
- The magnitude of the cask reactivity increase is slightly larger for the AO set than for the AFP set at ~1.0 pcm/K for AO and ~0.9 pcm/K for AFP. A bounding value in cask reactivity increase, regardless of the isotope set used, is 0.12% for every 100 K temperature increase.
- Use of the highest nodal-average fuel temperature in assembly depletion simulations will lead to conservative cask reactivity results.

b. <u>Bypass Water Density</u>

- Cask reactivity increases with decreasing bypass water density for both isotope sets. The magnitude of the cask reactivity increase is larger for the AO set than for the AFP set.
- The cask reactivity increase is less than 0.1% for every 1% reduction in bypass water density.

c. <u>Specific Power</u>

- Cask reactivity increases with increasing specific power for the AO isotope set, but the magnitude of the increase is very small: ~0.03% for a 10% specific power increase.
- The cask reactivity effect for the AFP isotope set is negligible.
- BWR BUC analyses crediting only the AO isotope set that model a high, bounding specific power during depletion result in highest cask reactivity. For analyses using the AFP isotope set, a reasonable specific power consistent with the expected depletion conditions produces statistically equivalent cask reactivity compared to higher specific powers.

d. Operating History

- Typical downtimes of ~30 days between cycles have a negligible impact on cask reactivity.
- Extended downtimes preceding an assembly's last irradiation cycle before being discharged lead to a decrease in cask k_{eff} values. The magnitude of the decrease is similar for the AO and AFP sets.
- Cask reactivity is negligibly affected by the power level during the assembly's last irradiation cycle relative to the lifetime-average power.
- An operating history with uniform power levels and no downtime between cycles provides a cask reactivity that is statistically equivalent with or higher than all other tested scenarios.

8. TASK 7: STUDY AND GUIDANCE FOR HANDLING CORRELATED FACTORS AFFECTING FUEL DEPLETION

The impacts of assembly-specific conditions were evaluated at 25 and 50 GWd/MTU assembly average burnups [10]. The assembly-specific conditions were studied for three different assemblies. The results obtained using these three assemblies are unlikely to bound all possibilities for all reactors. Additional research is needed to fully assess the impacts of assembly-specific conditions. The impact of modeling assembly-specific conditions that are correlated can provide cask reactivity reductions, but the magnitude of the reduction varies significantly with each fuel assembly.

- Cask reactivity is reduced by using assembly-specific conditions compared to limiting conditions. The magnitude of this reactivity reduction ranges from ~0.50% Δk_{eff} to more than 7% Δk_{eff}, depending on the assembly-specific conditions included.
- Using the assembly-specific burnup profile has the most significant impact on cask reactivity, which is consistent with previous findings. The impact varies significantly with the assembly selected.
- The impact of assembly-specific conditions on cask reactivity is greatest for assemblies with significant control blade insertion. Use of the control blade during operation changes the axial shape of the coolant density and burnup profile. Insertion of the control blade leads to less limiting coolant density and burnup axial profiles.
- The impacts of assembly-specific conditions on cask reactivity are greater for high-discharge burnups than for low-discharge burnups.

9. TASK 8: GUIDANCE FOR VALIDATION OF BWR BURNED FUEL ISOTOPIC COMPOSITION CALCULATIONS

Experimental data from the destructive assay of 77 BWR spent fuel samples have been evaluated to calculate margins for uncertainty in the predictions of isotopic inventories as applied to BUC criticality calculations [11]. Measurements cover a wide range of modern assemblies, including 8×8-2, 8×8-4, 9×9-7, GE11 9×9, 9×9-9, GE14 10×10, SVEA-96, and SVEA-100 10×10 designs. The data cover a range of void conditions up to 74% and a burnup range from 7 to 68 GWd/MTU. Most of the measurement data used are obtained from public references and additional information compiled and documented as part of the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) Spent Fuel Isotopic Composition (SFCOMPO) SNF measurement database. Several datasets used in this study are from proprietary programs. These data may be made available to support licensing applications through non-disclosure agreements.

The uncertainty analysis methodology used in this study is independent of the application model or the computational methods. The margins for isotopic uncertainty are developed by the direct application of measurement and calculated nuclide concentrations to the application model and statistical analysis of the results. This procedure, as applied to the major AO calculations, requires only minimal analysis of the isotopic distributions of individual nuclides since most samples include measurements for all major actinide isotopes. For minor actinide and FP credit, analysis of individual isotopic bias and uncertainty was used to develop surrogate isotope concentration data with uncertainties for isotopes not measured in a fuel sample.

This uncertainty analysis approach is demonstrated using SCALE 6.2.2 [6] with ENDF/B-VII.1 cross section data. Specifically, depletion calculations were performed using the Polaris code and criticality calculations were performed using KENO V.a.

A margin for uncertainty was developed from the 95% one-sided lower confidence limit (LCL) for the population of measurement data, and trending analysis was performed for sample burnup, average void, and k_{eff} of the application model. For the Polaris calculations and the GBC-68 application model, the k_{eff} bias for AO calculations was 253 pcm, with a conservative margin for isotopic uncertainty of 2170 pcm for the range

of fuel samples included. This margin was observed to be largely independent of fuel burnup or void fraction. For AFP calculations, the k_{eff} bias was 161 pcm, and the conservative margin used for isotopic uncertainty was 2,390 pcm.

These results are similar to those shown in previous studies for PWR burnup credit [17], where average biases for a dry storage cask of 320–720 pcm and uncertainties of 1,430–2,050 pcm were reported for fuel with a burnup less than 50 GWd/MTU. The larger uncertainties seen in the present study are likely attributed to the increased complexity of the BWR fuel, reactor modeling, and depletion conditions.

10. TASK 9: GUIDANCE FOR VALIDATION OF BWR BUC k_{eff} CALCULATIONS

A set of four application cases was defined to examine the number of available applicable benchmarks and to perform sample determinations of bias and bias uncertainty [12]. All four cases included spent fuel in a flooded GBC-68 cask. The fuel was depleted to 25 and 50 GWd/MTU, and the AO and AFP isotope sets were considered for both burnups. For the two cases with the AO isotope set, 172 experiments were identified as applicable at a burnup of 25 GWd/MTU, and 173 applicable experiments were identified for the 50 GWd/MTU case. A combination of LEU-COMP-THERM (LCT) and Haut Taux de Combustion (HTC) cases were identified as applicable at both burnups; nearly three times as many LCTs appeared to be applicable at 25 GWd/MTU than at 50 GWd/MTU. For the two cases with the AFP isotope set, 68 HTC cases were identified as applicable at 25 GWd/MTU, and 126 HTC cases were determined to be applicable at 50 GWd/MTU. For both isotope sets, more HTC cases were applicable at higher burnup because the fuel used in the HTC experiments was a closer match to higher burnup discharged fuel.

Bias and bias uncertainties were assessed for each of the four application models using trending and nontrending techniques. The trending techniques considered trends on EALF and on c_k. The nontrending biases and bias uncertainties were consistent, with the biases ranging from -0.00132 Δk to -0.00236 Δk, and the bias uncertainties ranging from 0.00530 Δk to 0.00672 Δk, with the highest values occurring for the 25 GWd/MTU AFP case. The trending analysis using EALF as the independent variable also showed consistent results, with biases ranging between 0 (best estimate value of +0.00044 Δk) and -0.00206 Δk, and the bias uncertainties ranging between 0.00646 and 0.00724 Δk. The trending analysis using c_k as the independent variable produced bias estimates ranging from -0.00047 to -0.00647 Δk, resulting from the variation of the slope of the trend line describing that data and the amount of extrapolation necessary to obtain a c_k of 1.0. The bias uncertainty results for the c_k trending analysis ranged from 0.00657 to 0.01556 Δk. The bias uncertainty results were influenced by the previously mentioned extrapolation distance and number of applicable experiments.

The identified potentially applicable critical benchmark experiments do not contain FPs or minor actinides (MAs), so a reactivity margin is needed to address the validation gap. The major actinide isotopes can be validated, so no additional margins are necessary for analyses using the AO isotope set. A reactivity margin of 2% of the FP and MA worth is likely appropriate for extended BWR BUC analyses that do not credit any residual gadolinium burnable absorber. A margin of 2.75% of the total FP and MA worth is applicable to models which do credit residual ^{155}Gd in BA rods along with FP ^{155}Gd.

11. SUMMARY

This paper summarizes work performed since late 2013 that examines several aspects of BWR BUC. Both peak reactivity and extended BWR BUC were studied. A list of the specific tasks performed in this project and the goals of each task are provided in Table I. The effects of many different parameters on peak reactivity BUC analysis are summarized in Table II for the AO isotope set and in Table III for the AFP isotope set. The results summarized in this report are intended to provide the necessary technical information for generation and regulatory review of applications crediting burnup for BWR SNF in transportation and/or storage systems.

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