Effects of Control Blade History, Axial Coolant Density Profiles, and Axial Burnup Profiles on BWR Burnup Credit

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ABSTRACT

A technical basis for peak reactivity boiling water reactor (BWR) burnup credit (BUC) methods was recently generated, and the technical basis for extended BWR BUC beyond peak reactivity is now being developed. In this paper, a number of effects related to extended BWR BUC are analyzed, including three major operational effects in BWRs: the coolant density axial distribution, the use of control blades during operation, and the axial burnup profile. Specifically, uniform axial coolant density profiles are analyzed and compared to results for realistic coolant density profiles taken from operating data. A new temporal fidelity study was performed for the coolant density profile change as a function of operating time by consecutively simulating the coolant density profiles first–, second–, and third–cycle fuel assemblies. Realistic control blade histories and their impact on cask reactivity are also analyzed. Preliminary analysis of the axial burnup profile is also provided.

Key Words: **BWR, burnup credit, storage, transportation, spent nuclear fuel**

1. INTRODUCTION

Oak Ridge National Laboratory (ORNL) and the US Nuclear Regulatory Commission (NRC) have undertaken a multiyear project to investigate the application of burnup credit (BUC) for boiling water

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reactor (BWR) fuel in storage and transportation systems (often referred to as casks) and spent fuel pools (SFPs). This work is divided into two main phases. The first phase investigated the applicability of peak reactivity methods currently used in SFPs to transportation and storage casks and the validation of reactivity calculations and spent fuel compositions within these methods [1, 2]. The second phase focuses on extending BUC beyond peak reactivity. This paper documents the analysis of the effects of control blade insertion history, along with moderator density and burnup axial profiles for extended BWR BUC.

Extended BUC has been applied for pressurized water reactor (PWR) SFPs and storage and transportation casks for a number of years. Peak reactivity analyses have been performed for storage of BWR fuel in SFPs over the last 20 years. In this context, *extended BUC* refers to BWR fuel beyond peak reactivity burnup values, typically greater than 15 to 20 GWd/MTU, but the exact burnup point at which a licensee would use peak reactivity methods or extended BUC methods has yet to be defined. The existence of two separate methods of BUC in BWRs is associated with the gadolinia burnable absorber used in BWRs. Because of the burnable absorber, the reactivity of the fuel assembly increases from the beginning of irradiation until the gadolinium has been depleted and the reactivity peaks. After the peak, the reactivity of the fuel decreases for the remainder of operation.

To apply full BUC to BWRs, a number of effects need to be analyzed. The operation of BWRs is significantly different from that of PWRs. This paper analyzes three major operational effects in BWRs: the coolant density axial distribution, the use of control blades during operation, and the axial burnup profile. Although some results originally published in Ade et al. [3] are discussed, the main focus of this paper is additional studies performed for axial coolant density profile and control blade history, and the initial results for the study of the axial burnup profile.

In addition to previously generated results [3], an axial moderator density profile was constructed using data from three separate fuel assemblies in order to simulate a realistic moderator density history for a fuel assembly. Realistic control blade histories were extracted from the operating data, simulated, and then compared to the hypothetical control blade histories. Finally, the initial burnup profile results were generated using every end-of-cycle (EOC) burnup profile in the considered operating data.

2. METHODOLOGY

The SCALE code system [4] was used for all calculations. Either SCALE/TRITON or STARBUCS was used for depletion calculations, whereas CSAS/KENO V.a was used for spent fuel cask criticality calculations. In the study of the axial coolant density profile and the axial burnup profile, STARBUCS was used extensively to generate depleted fuel isotopics. However, for the control blade history studies, SCALE/TRITON was required for depletion calculations due to current limitations in the STARBUCS methodology. For various parts of these studies, physical operating data, including core simulator-generated core-follow data, were used to simulate realistic operating conditions.

TRITON is a multipurpose SCALE control module for transport, depletion, and sensitivity and uncertainty analysis [5]. In the context of this work, TRITON was used to perform 2-D depletion calculations with time-dependent moderator density and control blade insertion state, as well as to generate ORIGEN libraries for follow-on calculations in STARBUCS. STARBUCS was used to perform depletion calculations and generate depleted fuel isotopics for the moderator density profile study, as well as the axial burnup distribution study. The depleted fuel isotopics from TRITON and STARBUCS were decayed for five years, and then SCALE material specifications were generated from these calculations and passed to the CSAS/KENO V.a fuel cask model.

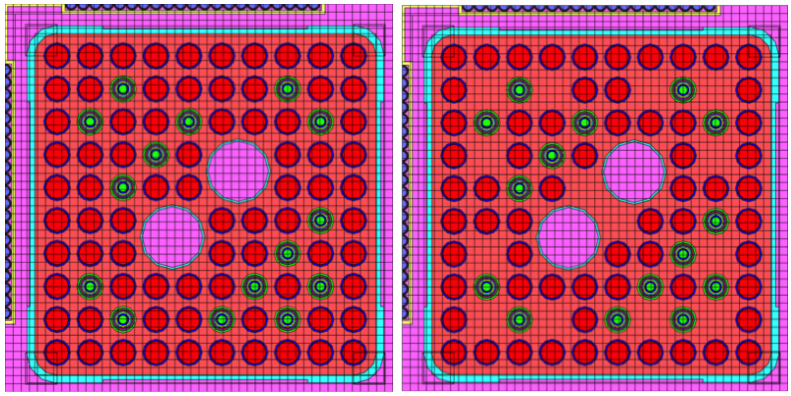
The Criticality Safety Analysis Sequence CSAS/KENO V.a in SCALE 6.1.3 was used for reactivity calculations for the generic burnup credit 68-assembly fuel cask computational benchmark model (GBC-68) [6]. The sequence provides automated problem-dependent cross section processing followed by 3D multigroup Monte Carlo neutron transport calculations to solve the k-eigenvalue problem. All calculations were performed using the 238-group neutron cross section library based on ENDF/B-VII.0 data.

One fuel assembly design type, GE14 (10×10), was used in these studies. In order to model axial changes in the parameters of interest, the full-length GE14 fuel assembly was modeled as 25 uniformly-sized 6-inch axial regions (corresponding to the regions used in the core-follow data). Fuel depletion calculations were performed for each of these 25 axial regions. Each region is modeled as a two-dimensional slice in either TRITON or STARBUCS, and the conditions in each of the slices corresponds to the axial moderator density profile, axial burnup profile, and control blade history being simulated.

Two different sets of nuclides are considered for fuel modeling in the CSAS/KENO V.a models: (1) major actinides only (AO) and (2) major and minor actinides and major fission products (AFP). The nuclides used in the AO and AFP nuclide sets are taken from NUREG/CR-7109 [7] and are the same as those typically used when calculating PWR BUC. The AO isotopes contain uranium and plutonium isotopes, as well as ^{241}Am and ^{16}O. The AFP isotope set contains all the isotopes in the AO set, ^{237}Np, ^{243}Am, as well as 16 fission products important to reactivity.

Due to the presence of part-length fuel rods which terminate at approximately half the total height of the fuel assembly, the GE14 fuel assembly contains two primary axial zones (or levels). These two axial regions are the dominant (DOM or full) and vanished (VAN) lattices. TRITON representations of the DOM and VAN lattices are shown in Figure 1. All gadolinium-bearing rods contain the same absorber loading in both the DOM and VAN lattices. The fuel assembly design used in these studies is based on an actual assembly from the detailed core follow data considered for this research. All TRITON depletion calculations used 4.5 wt% ^{235}U in all fuel rods including the 15 rods containing 7wt% Gd_2O_3. This single assembly design is used throughout the calculations to assess the effect of the different axial moderator density distributions, control blade histories, and axial burnup profiles independent of any effects potentially caused by differing fuel or absorber loadings.

Each parameter being studied was varied singly to isolate its effect on fuel cask reactivity. All other



(a) Full lattice (b) Vanished lattice

Figure 1. TRITON GE14 model representations. The red colored rods contain only uranium, while green rods contain uranium and gadolinium burnable absorber.

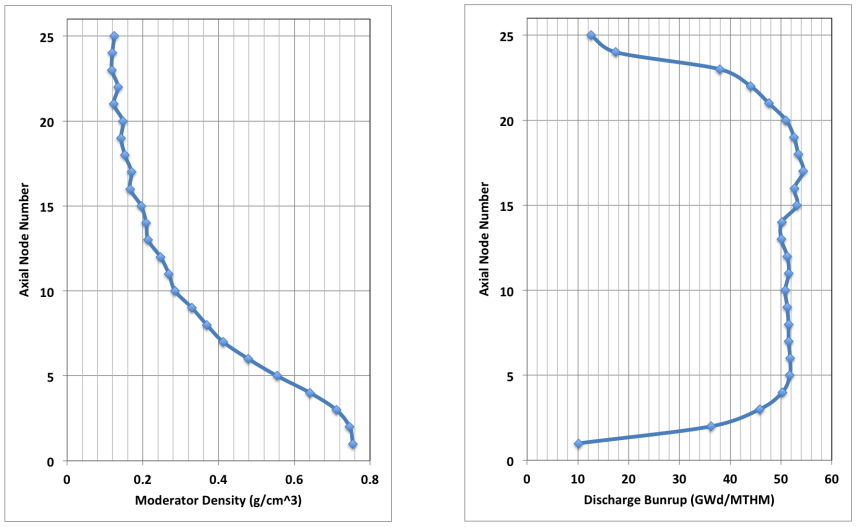
parameters, including specific power, fuel temperature, and moderator temperatures, are unchanged throughout these studies. The axial moderator density profile and axial burnup profile used in the studies discussed here are shown in Figure 2. These distributions are taken from similar fuel assemblies and are expected to be near-limiting in regard to cask reactivity. In general, the distributions shown in Figure 2 are used in the studies unless that parameter is the one being varied. The control blades are modeled as removed for all but the control blade usage study.

3. RESULTS

3.1. Axial Coolant Density (Void Fraction) Profile

After initial axial coolant density (or void fraction) profile studies were performed [3], additional calculations were completed using a number of different uniform coolant density axial profiles. These profiles were modeled to ascertain the level of penalty/conservatism that would result from using a uniform low-density coolant profile over the entire irradiation. Previous studies simulated different axial profiles (named Profiles 1–10), including those taken from the core operating data [3]. The "Min" profile corresponds to the minimum void fraction in each axial node of ten real profiles. The "Avg" profile is the average over all fuel assemblies in the core and over the whole cycle of operating time. The uniform profiles (40%, 89.1%, 89.2%, 89.5% and 90.2% void fraction) use a uniform moderator density axial profile corresponding to that particular void fraction. The uniform profiles were added to evaluate simplifying assumptions that analysts may choose when performing burnup credit analysis, and the penalty (or lack thereof) of those assumptions.

The cask reactivity values for the actinide and fission product isotope sets can be found in Table I for assembly–average burnup values of 30, 40, and 50 GWd/MTHM. The difference between the most lim-



(a) Axial moderator density profile. (b) Axial discharged burnup profile.

Figure 2. Reference axial moderator density and axial burnup profiles utilized.

iting realistic axial void profile (Profile 1) and every other profile has been calculated and included in Table I. The uniform 40% void profile, which is commonly used as a core-average void fraction value, results in nonconservative cask reactivities from –4.8 to –7.3% Δk_{eff} for assembly-average burnup between 30 and 50 GWd/MTHM. For this reason, the uniform 40% void profile is not recommended for BWR BUC analyses. The core-average moderator density profile results in nonconservative cask reactivities from –2.0 to –2.9% Δk_{eff} for assembly-average burnup between 30 and 50 GWd/MTHM. The Min profile is less than 0.1%Δk_{eff} conservative for assembly-average burnup between 30 and 50 GWd/MTHM. The uniform high-void profiles from 89.1% to 90.2% void are all conservative by 0.6–1.1% Δk_{eff} , depending on the profile and burnup point. These results indicated that using a uniform density profile that corresponds to a very low moderator density will result in conservative cask reactivity estimations.

The aforementioned studies considered a single moderator density profile for each of the three consecutive operating cycles; i.e., a realistic profile is modeled, but that profile is kept constant for each cycle. This assumption is clearly nonphysical; near the end of the assembly life, the axial void fraction profile is likely to have a relatively low average void fraction because of the limited amount of power produced by the assembly. However, the logic behind this assumption is that any limiting axial void fraction profile repeated for three consecutive cycles will be more limiting than the void fraction profile experienced by a single assembly for any three cumulative real cycles of operation.

To test the hypothesis that the use of a single moderator density profile repeated for multiple cycles is conservative relative to more realistic operation, an additional time-dependent moderator density profile was constructed from the available operating data. This time-dependent profile was constructed using the moderator density profile from three different fuel assemblies (fresh, once- and twice-burned), herein referred to as the three-cycle constructed profile (3CCP). This particular study is very similar

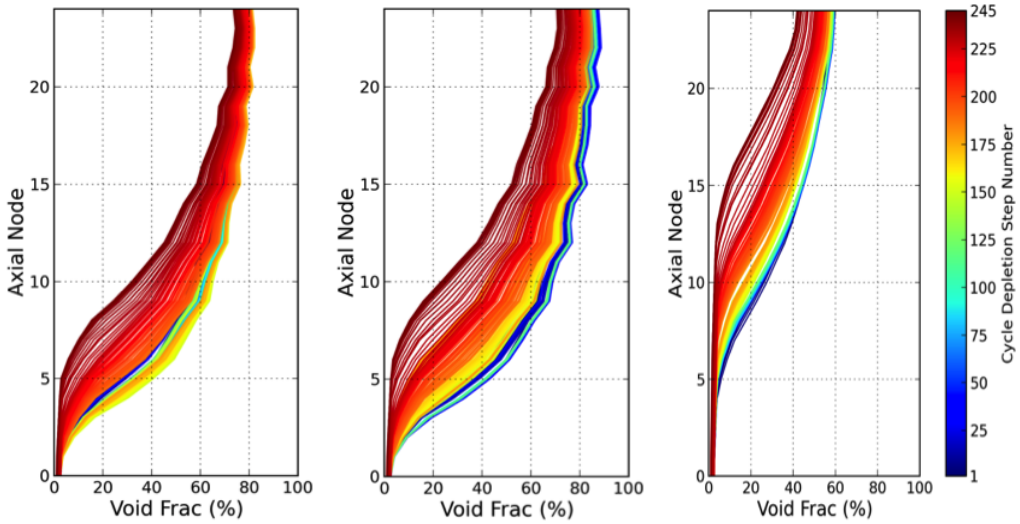
Table I. Calculated cask k_{eff} values for the each axial void profile at selected burnup points

	30 GWd/MTU		40 GWd/MTU		50 GWd/MTU	
Profile Name	k_{eff}	$\Delta^{\dagger}$	k_{eff}	$\Delta^{\dagger}$	k_{eff}	$\Delta^{\dagger}$
Profile 1	0.86367	–	0.84389	–	0.82709	–
Profile 2	0.86335	−0.00032	0.84344	−0.00045	0.82645	−0.00064
Profile 3	0.86336	−0.00031	0.84304	−0.00085	0.82614	−0.00095
Profile 4	0.86329	−0.00038	0.84299	−0.00090	0.82639	−0.00070
Profile 5	0.86304	−0.00063	0.84285	−0.00104	0.82614	−0.00095
Profile 6	0.86293	−0.00074	0.84260	−0.00129	0.82583	−0.00126
Profile 7	0.86230	−0.00137	0.84211	−0.00178	0.82528	−0.00181
Profile 8	0.86207	−0.00160	0.84178	−0.00211	0.82485	−0.00224
Profile 9	0.86196	−0.00171	0.84157	−0.00232	0.82476	−0.00233
Profile 10	0.86206	−0.00161	0.84158	−0.00231	0.82473	−0.00236
Min.	0.86438	0.00071	0.84478	0.00089	0.82848	0.00139
Avg.	0.84364	−0.02003	0.81910	−0.02479	0.79799	−0.02910
40% Void	0.81541	−0.04826	0.78298	−0.06091	0.75351	−0.07358
89.1% Void	0.86993	0.00626	0.85040	0.00651	0.83412	0.00703
89.2% Void	0.86998	0.00631	0.85067	0.00678	0.83451	0.00742
89.5% Void	0.87066	0.00699	0.85166	0.00766	0.83549	0.00840
90.2% Void	0.87235	0.00868	0.85341	0.00952	0.83766	0.01057

† Δk_{eff} calculated with respect to the most reactive real profile, Profile 1.

to the temporal fidelity study conducted by Ade et al. [3], however the previous study used the moderator density profile for a single assembly rather than for three profiles, each taken from a different assembly. The previous temporal fidelity study was used to determine the number of steps (temporal fidelity) needed to accurately capture the time-dependent change in the axial moderator density profile as a function of depletion, which indicated that usage of a cycle-average moderator density is suitable for modeling purposes. Somewhat contrary to these results, associated work [8] showed rather large impacts on criticality when averaging moderator density over a particular fuel assembly lifetime, so an additional study was warranted. It is important to note that the fuel assembly in Martinez et al. [8] was irradiated for five cycles and the current study considers three cycles of operation.

The axial void profiles selected for the current study can be found in Figure 3. Axial void profiles for three different fuel assemblies were chosen for analysis. In each of the plots in Figure 3, each line represents one state point of the more than 200 state points in the nodal simulator calculation. The lines in Figure 3 are colored by the step number; blue colored lines are near the beginning of the cycle (BOC), while red lines are near the end of the cycle (EOC). Figure 3(a) shows the variation in void fractions for a fresh fuel assembly. The void profile is relatively low at BOC (blue lines are covered by red lines); the void fraction increases as gadolinium in the fuel assembly is depleted, after which it decreases until EOC. In the once- and twice-burned fuel shown in Figures 3(b) and 3(c), the void fraction simply moves from higher values to lower values throughout the cycle from BOC to EOC.



(a) fresh fuel (b) once-burned fuel (c) twice-burned fuel

Figure 3. Three void profiles used in the 3-cycle constructed moderator density profile.

As in the previous temporal fidelity study [3], the full detail of these three cycles was approximated by averaging the time-dependent void fraction into 25 steps per cycle. In this study, the 25-step approximation of void fraction (moderator density) is considered the reference solution. Two other solutions were generated that correspond to one step per cycle (cycle-average void fraction) and lifetime-average (void fraction averaged over all three cycles). A figure depicting the three different moderator density treatments can be found in Figure 4, which shows the moderator density variation for the third node from the top of the assembly.

In addition to the three-cycle constructed profile, the individual moderator density profiles were simulated individually by repeating the moderator density profile for three consecutive cycles in order to compare cask criticality values to the 3CCP. The cask criticality results for the 25-step, cycle-average, and lifetime-average void fraction treatments can be found in Table II. The lifetime-average results for the fresh, once burned, and twice burned fuel are omitted from the table, as the results would be identical to the cycle-average results. Because the same cycle is repeated three times, cycle-average and lifetime-average are identical simulations.

The 3CCP results shown in Table II indicate that the lifetime-average approach is conservative when compared to the 25-step and cycle-average approaches by 0.185 and 0.179% Δk_{eff} for AO and by 0.101 and 0.112% Δk_{eff} for AFP, respectively. The higher reactivity of the lifetime-average cases compared to the more detailed temporal modeling is expected due to the change in moderator density as a function of time. As shown in Figure 4, the lifetime-average moderator density is lower than that of the twice burned fuel assembly, so more plutonium is produced near the end of life in the lifetime-average case than in the 25-step and cycle-average cases. Also, as expected from the moderator density trends in Figure 4, the 3CCP criticality results fall between those for the twice burned fuel moderator density profile and the fresh and once burned fuel assembly moderator density profiles. These results confirm

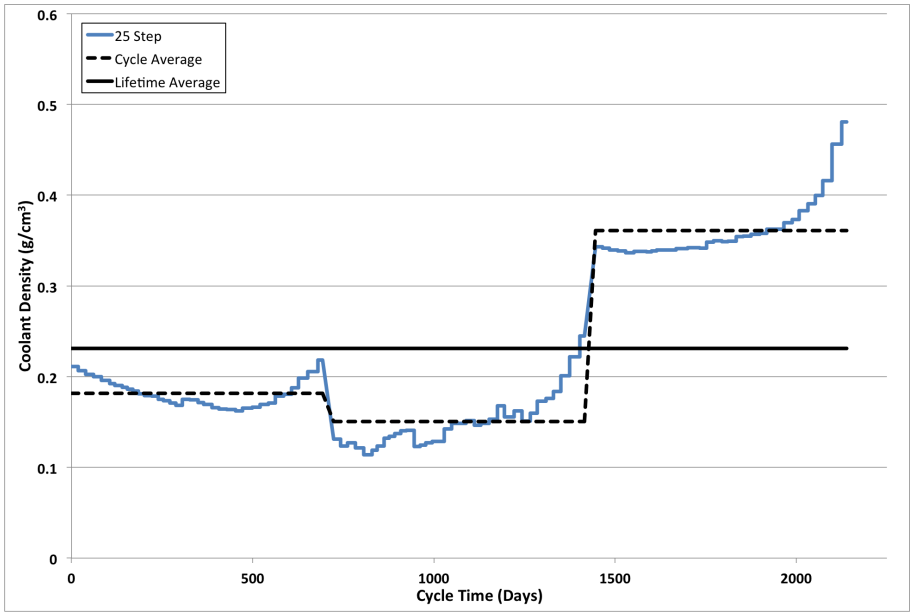


Figure 4. Moderator density change in node 3 (from the top) as a function of time for the three temporal averaging approaches.

the hypothesis that any *limiting* moderator density profile simulated for three consecutive cycles is more conservative than the actual moderator density profile typically observed in BWR fuel assemblies.

3.2. Control Blade Histories

Previous work [3] analyzed a number of hypothetical control blade histories which contain very simplistic control blade patterns. These included control blade histories such as blades inserted for the entire irradiation, blades removed for the entire irradiation, blades inserted for the first half of irradiation, and so on. Through analysis of the hypothetical histories, key k_{eff} sensitivities—such as control

Table II. Cask criticality results for the three-cycle constructed profile and the individual moderator density profiles

Isotope Set	Temporal Fidelity	3CCP	Fresh	Once Burned	Twice Burned
	25-step	0.80609	0.81119	0.81358	0.79966
AO	Cycle-average	0.80615	0.81122	0.81368	0.79974
	Lifetime-average	0.80794	—	—	—
	25-step	0.72266	0.72445	0.72498	0.72202
AFP	Cycle-average	0.72256	0.72458	0.72509	0.72217
	Lifetime-average	0.72368	—	—	—

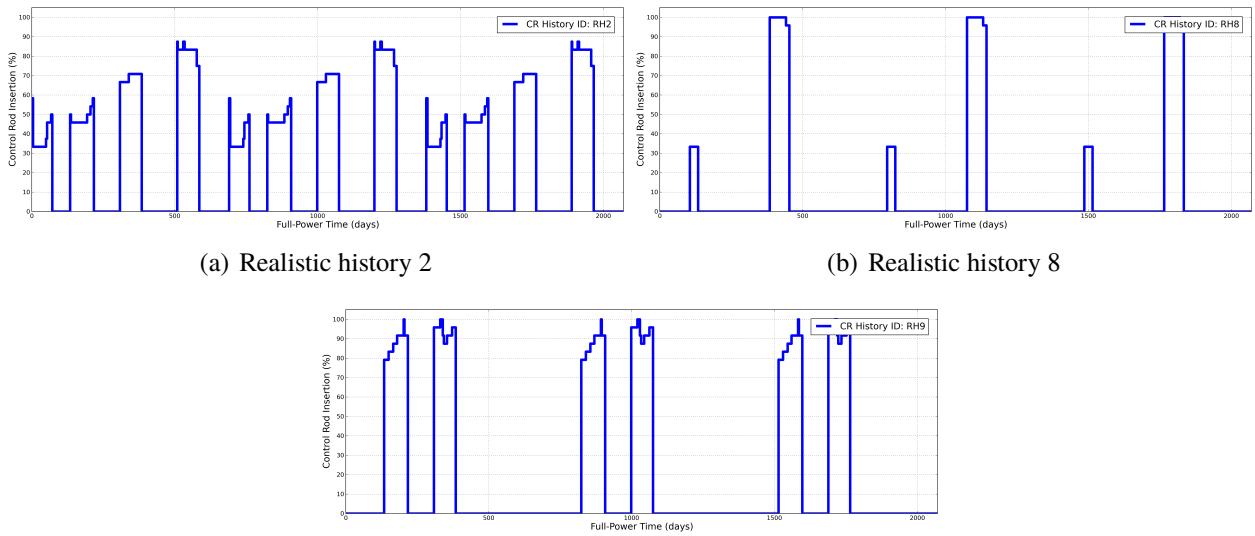
blade insertion depth (or elevation), control blade insertion time (early or late in life), and control blade insertion duration—were analyzed.

This study revealed that the fuel cask reactivity is highly sensitive to control blade insertion depth. This sensitivity was further analyzed by running a number of calculations with the control blades inserted to depths between 50 and 100% for all three consecutive cycles. These calculations showed that the sensitivity to control blade insertion becomes very high when the control blade reaches the top four nodes in the fuel assembly. That is, the cask reactivity increases nonlinearly as a function of control blade insertion depth in the top four nodes of the fuel assembly. This result highlights the end effect in which the vast majority of fissions in the cask geometry occur in the uppermost nodes of the assembly. This occurs because of the relatively low burnup (high residual ^{235}U) fuel in the upper portion of the fuel assembly, combined with the relatively low moderator density (increase plutonium production) in these regions during depletion and the increased moderation in the axial ends of spent fuel storage and transportation casks.

To compare the hypothetical histories to more realistic histories, control blade insertion data were extracted from the available operating data. The control blade insertion depths were then plotted, and certain histories were selected for analysis. With the hypothetical history cases indicating that the upper nodes have a disproportionate effect on cask reactivity compared to lower nodes, ten realistic control blade histories were extracted from the operating data and simulated. Three key histories of the ten selected realistic control blade histories can be found in Figure 5. The additional realistic histories are omitted for brevity. However, the ten chosen histories correspond to locations in which control blades were inserted very deeply into the core or were inserted for extended periods of operation. As in previous studies, the single-cycle control blade history was repeated for three consecutive cycles.

Because only one cycle of operating data is available, the full-length control blade histories were constructed by simulating a selected single-cycle control blade history for three consecutive cycles. Fuel assemblies are unlikely to experience significant operational periods with control blades inserted for three consecutive cycles due to fuel assembly shuffling and decreasing fuel reactivity with increasing burnup. In general, fresh fuel assemblies are placed relatively close to the center of the core and then moved to the radial periphery of the core. The control blades relatively close to the center of the core are typically used frequently, as they have the greatest effect on core reactivity, while the control blades near the periphery are not frequently used during operation. In addition, control blades are not typically used significantly at EOC due to the reduction of total core reactivity as a result of fuel depletion. This cycle-by-cycle movement of fuel assemblies, preferential use of certain control blade locations, and limited use of control blades at EOC are likely to lead to fuel assemblies that experience more control blade insertion near BOL and less control blade insertion near end of life (EOL). As such, the simulation of three cycles with large control blade histories should produce conservative criticality results when compared to typical operation.

After each of the ten realistic histories had been modeled using TRITON, the EOL isotopics were decayed for five years and inserted into the fuel cask criticality model. The cask criticality results for histories RH1–RH10 can be found in Table III. The total spread (highest minus lowest) of realistic



(a) Realistic history 2

(b) Realistic history 8

(c) Realistic history 9

Figure 5. Selected realisitc control blade histories.

profile cask criticality data for the AO and AFP isotope sets is 0.49% and 0.59%Δk_{eff} , respectively. Initially, one might expect that the RH2 history, which includes the highest integral control blade history, would also produce the highest cask reactivity. Surprisingly, RH2 has the seventh highest cask reactivity for both the AO and AFP isotope sets. This occurs because, although RH2 has the most frequent control blade usage, it contains only one instance per cycle when the control blade is inserted to a depth greater than 80% into the core. By contrast, RH9, which produces the highest cask reactivity for both the AO and AFP isotope sets, has two relatively lengthy control blade insertions, both at depths greater than 90% into the core. Two clear conclusions can be drawn from these results. Because of the high sensitivity to very deep control blade insertion, histories that have control blade insertion greater than 80% into the core result in a noticeable impact on cask reactivity. By contrast, cases with control blade insertions less than 50% of the fuel length are virtually indiscernible (within the statistical deviation) from the base case with control blades fully withdrawn.

The most limiting realistic history (RH9) differs from the most limiting hypothetical history (rods inserted at or near full depth for the entire irradiation) by –3.66% and –3.68%Δk_{eff} for the AO and the AFP isotope sets, respectively. The biases between RH9 and the most limiting hypothetical histories indicate that the assumption of fully inserted rods for the entire irradiation is an overly conservative assumption. Comparing RH9 to the case in which the control blades are fully withdrawn shows differences of 0.49% and 0.59%Δk_{eff} for the AO and AFP isotope sets, respectively. This difference indicates that neglecting control blade insertion during depletion could result in nonconservative cask criticality values up to 0.6%Δk_{eff} . However, this difference should not be viewed as a bounding estimate of the effect because the result is based on only a single cycle of control blade histories. Additional data from other plants and for the entire assembly life would be needed to generate a more generically applicable estimate of the impact of realistic control blade usage on cask reactivity .

Table III. Cask k_{eff} values for the AO and AFP isotopes sets

AO isotope set		**AFP isotope set**	
History	k_{eff} †	History	k_{eff} †
RH9	0.83733	RH9	0.73781
RH8	0.83627	RH6	0.73583
RH3	0.83610	RH8	0.73533
RH6	0.83569	RH3	0.73532
RH10	0.83440	RH10	0.73397
RH5	0.83304	RH5	0.73300
RH2	0.83278	RH2	0.73281
RH1	0.83258	RH7	0.73248
RH4	0.83253	RH4	0.73210
RH7	0.83247	RH1	0.73188

† All k_{eff} calculations have estimated standard deviations ≤ 0.0001.

3.3. Axial Burnup Profile

No BWR database currently exists that is analogous to the PWR database used in Wagner, DeHart, and Parks [9], so a set of profiles was generated. For this study, EOC burnup profiles were generated for all 624 assemblies in the core follow data set used in previous studies and elsewhere [1–3]. These profiles were normalized to enable comparisons of profiles that are independent of the burnups of the assemblies from which the profiles were taken. Representative profiles from low burnup assemblies are shown in Figure 6(a), intermediate burnup in Figure 6(b), and relatively high burnup in Figure 6(c). The top of the assembly in Figure 6(c) corresponds to low node numbers (i.e., node 1 is at the top of the assembly, and node 25 is at the bottom).

There are two general profile shapes in the low burnup range (less than 25 GWd/MTU), as shown in Figure 6(a). The flatter profiles show evidence of control blade insertion, as the burnup in the lower portion of these assemblies is suppressed relative to the profiles without control blade insertion, which manifest a much more bottom skewed profile. The profiles resulting from assemblies with control blade insertion are indicated by a dashed line, and the others are indicated by a solid line. The variation among profiles is dramatic; the difference between the maximum and minimum relative burnup at node 5 (relative to the top of the assembly) in the profiles shown is approximately 0.3 (i.e., greater than 30%).

Figure 6(b) shows normalized burnup profiles for assemblies that have experienced multiple cycles of irradiation and have an EOC burnup between 25 and 40 GWd/MTU. In general, the burnup profiles are more similar in this burnup range than the first cycle profiles shown in Figure 6(a). One profile shows a more bottom skewed shape than the others, as indicated with the dashed line. This feature, nicknamed the "hockey stick," is present for many more assemblies in the high burnup profiles shown in Figure

6(c) and is related to usage of a 12-inch top axial blanket rather than a 6-inch top axial blanket that is used for many assemblies.

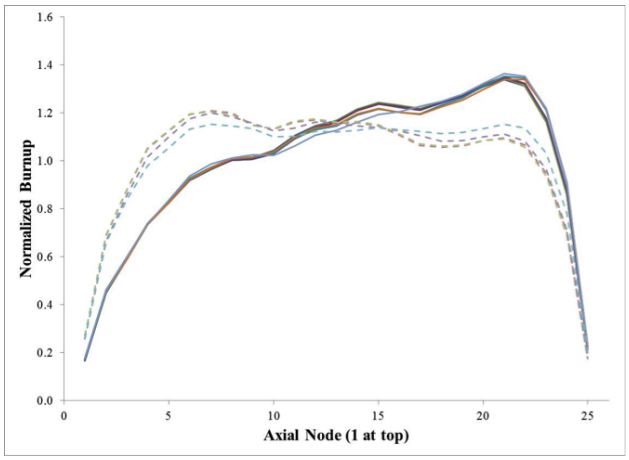
The BWR SNF in the domestic inventory typically has natural enrichment axial blankets. A routine assumption made in PWR BUC is that no blankets are present, and this assumption is conservative because it significantly increases the quantity of fissile material in the relatively high reactivity ends of the assembly. The assembly ends have high reactivity because they have low burnup; this effect was studied extensively for PWR SNF in Wagner, DeHart, and Parks [9]. The examination of the effects of BWR axial burnup profiles thus starts with the same assumption: that the axial blankets are not present in the discharged fuel assemblies. These profiles were generated by a 3D core simulator model that appropriately represented the fuel as manufactured, (i.e., with blankets), but the GBC-68 cask model introduces a modeling simplification of full enrichment fuel over the entire length of the assembly. The GE14 assembly model includes the full and vanished lattices, but it does not include a reduced enrichment section for the axial blankets.

Cask criticality results were generated for all 624 axial burnup profiles from the available core follow data at assembly-average burnup values of 30, 40, and 50 GWd/MTU, however, only a summary of those calculations is presented here. These models include the modeling simplification of full-length enriched fuel, and are performed for both the AO and AFP isotope sets. A uniform axial burnup profile that is depleted with the same depletion conditions and axial moderator distribution as the nonuniform profiles is included for comparison. A summary of these calculations (cask k_{eff} values) can be found in Table IV.

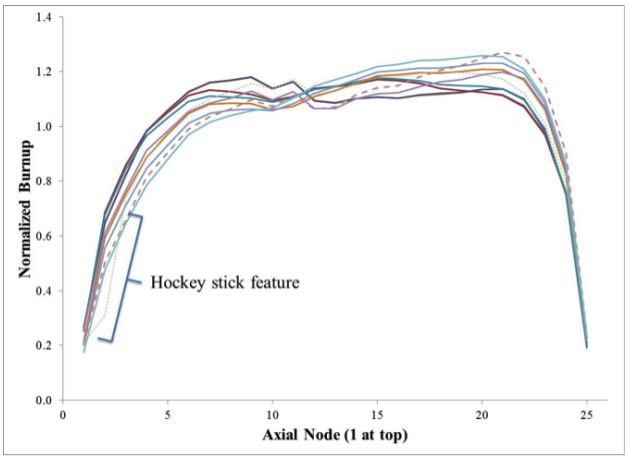
Table IV shows the minimum, average, and maximum k_{eff} values for each burnup for the AO and AFP isotope sets, as well as the standard deviation of the k_{eff} values for each distribution. Table IV also shows the results for the uniform burnup profiles. The values presented in the table show that a wide range (difference between maximum and minimum) of cask k_{eff} values result from the 624 axial burnup profiles analyzed at each of the three considered burnups and that the range increases with burnup. For the AO isotope set, the range is approximately 3.3% Δk_{eff} at 30 GWD/MTU, and it increases to almost 5% Δk_{eff} at 50 GWd/MTU. For the AFP isotope set, the range is 4.0% Δk_{eff} at 30 GWd/MTU, and it increases to 6.1% Δk_{eff} at 50 GWd/MTU. The range of cask k_{eff} values is relatively constant in terms of standard deviations, with all six distributions (two isotope sets at each of three burnups) having a width between 3.35 and 3.77 standard deviations.

The difference between cask k_{eff} calculated using a distributed burnup profile and that calculated using a uniform profile is called the *end effect*. When the end effect is positive, modeling the distributed (more realistic shaped) burnup profile is more conservative. Comparing the uniform profile to the minimum, maximum, and average results in Table IV, it is clear that all simulated profiles exhibit the end effect at all tested burnup values.

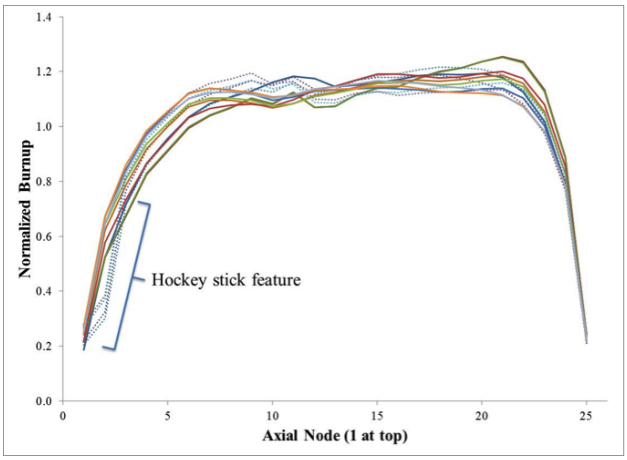
The end effect values for the AO isotope set increase with burnup as expected, and all 624 profiles have a positive end effect by 30 GWd/MTU. This is a somewhat surprising result, but it is strongly influenced by the modeling simplification made by using fully enriched fuel in place of natural blankets. At 30



(a) Selected normalized low burnup profiles



(b) Selected normalized medium burnup profiles



(c) Selected normalized high burnup profiles

Figure 6. Selected normalized burnup profiles for fuel assemblies with EOC burnups less than 25 GWd/MTU (low burnup), between 25 and 40 GWd/MTU (medium burnup), greater than 40 GWd/MTU (hight burnup)

Table IV. Cask k_{eff} distribution data for the unblanketed fuel models for assembly-average burnup values of 30, 40, and 50 GWd/MTHM. The uniform burnup profiles k_{eff} results are also provided.

	AO Isotope Set			**AFP Isotope Set**		
	30	40	50	30	40	50
Uniform Burnup Profile	0.85417	0.80460	0.75602	0.79371	0.73231	0.67350
Minimum k_{eff}	0.85722	0.82286	0.79255	0.81010	0.77259	0.73988
Maximum k_{eff}	0.89052	0.86527	0.84241	0.85027	0.82220	0.80089
Average k_{eff}	0.87505	0.84535	0.81854	0.83154	0.79842	0.76911
Standard deviation	0.00883	0.01201	0.01477	0.01132	0.01481	0.01763
Range (max-min)	0.03330	0.04241	0.04986	0.04017	0.04961	0.06101

GWd/MTU, the end effect varies from 0.3% Δk_{eff} to 3.6%Δk_{eff}; at 50 GWd/MTU, the minimum end effect increases to 3.7% Δk_{eff}, and the maximum value is 8.6% Δk_{eff}. The end effect values for the AFP isotopes set also increase with burnup as expected. As with the AO isotope set, all profiles result in positive end effects by 30 GWd/MTU. The magnitude of the end effects is significantly greater for the AFP set than for the AO set. The end effect ranges from 1.6% Δk_{eff} to 5.7% Δk_{eff} at 30 GWd/MTU and from 6.6% Δk_{eff} to 12.7% Δk_{eff} at 50 GWd/MTU. These numbers dwarf the results presented in Wagner, DeHart, and Parks [9], but the assemblies considered in that work were exclusively unblanketed fuel.

These end effect values are quite large, and extrapolation to lower burnups indicate that positive end effects may exist below 20 GWd/MTU. Positive end effects in this burnup range could impact peak reactivity methods, such as those described in Marshall et al. [2] since those methods typically assume a two-dimensional planar representation of the fuel assembly. A positive end effect would indicate that this approach may be nonconservative without considering distributed burnup profile effects. The burnup profiles that exist in the peak reactivity burnup range of 7–20 GWd/MTU might be significantly different than EOC profiles. The residual gadolinium burnable absorber (BA) may also impact the results at these lower burnups. Residual BA concentrations are much lower by EOC and are therefore not a significant contribution to cask reactivity. Additional work is needed to examine this possibility, as the burnup profiles used in this study are all EOC profiles.

4. Conclusions

In this paper, a number of effects related to extended BWR BUC were analyzed, including three major operational effects in BWRs: the coolant density axial distribution, the use of control blades during operation, and the axial burnup profile. Building on previous work [3], uniform axial moderator density profiles were analyzed and an additional temporal fidelity study was performed combining moderator density profiles for three different fuel assemblies. Realistic control blade histories were simulated and cask criticality results were compared to previously-generated constructed control blade histories. A

preliminary study of the axial burnup profile was also performed.

The axial moderator density study, indicated that the axial moderator density profiles can have a significant impact on calculated cask k_{eff} and must be treated appropriately to ensure conservative analysis results. While the set of profiles analyzed here is somewhat limited, being based on only one cycle of operation, important conclusions can still be drawn from these studies. A cycle-averaged moderator density can be used in each node of an axial moderator density profile for depletion calculations with an appropriate penalty for conservatism. Limiting profiles are those with low moderator densities in the top nodes of the assembly. It is also clear from the three cycle constructed profile study that using a single limiting moderator density profile for multiple cycles will be more conservative than using the varying moderator density profile experienced by actual fuel assemblies.

The control blade usage study, shows that control blade usage can impact cask k_{eff} and must be treated appropriately to ensure conservative analysis results. The impact is less severe than original expectations, given that the control blades need to be inserted more than 75% into the core before having a noticeable effect. Control blade insertion to depths of 50% or less has virtually no impact on cask reactivity; deeper control blade insertions (80% or more) have a much greater impact than frequent, shallower insertions. Fuel assemblies are unlikely to experience significant operational periods with control blade insertion late in life due to lower reactivity and placement near the core periphery. Therefore, the limiting cases that simulated heavy control blade insertion for three consecutive cycles may be overly conservative compared to actual operation.

The axial burnup profiles can also have a significant impact on cask k_{eff} and must be treated appropriately to ensure conservative analysis results. The span (difference between maximum and minimum) of k_{eff} results increases with increasing discharge burnup. Comparison of the uniform and distributed burnup profiles indicates that realistic, distributed burnup profiles must be considered for extended BWR BUC. All profiles tested in this study show a positive end effect.

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