COOLANT DENSITY AND CONTROL BLADE HISTORY EFFECTS IN EXTENDED BWR BURNUP CREDIT

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ABSTRACT

Oak Ridge National Laboratory and the US Nuclear Regulatory Commission have initiated a multiyear project to investigate the application of burnup credit (BUC) for boiling water reactor (BWR) fuel in storage and transportation casks. This project includes two phases. The first phase investigates the applicability of peak reactivity methods currently used for spent fuel pools to spent fuel storage and transportation casks and the validation of reactivity (k_{eff}) calculations and predicted spent fuel compositions. The second phase focuses on extending BUC beyond peak reactivity. This paper documents work performed to date investigating some aspects of extended BUC. The technical basis for application of peak reactivity methods to BWR fuel in storage and transportation systems is presented in a companion paper.

Two reactor operating parameters are being evaluated to establish an adequate basis for extended BWR BUC: (1) the effect of axial void profile and (2) the effect of control blade utilization during operation. A detailed analysis of core simulator data for one cycle of a modern operating BWR plant was performed to determine the range of void profiles and the variability of the profile experienced during irradiation. Although a single cycle does not provide complete data, the data obtained are sufficient to determine the primary effects and to identify conservative modeling approaches. These data were used in a study of the effect of axial void profile. The first stage of the study was determination of the necessary moderator density temporal fidelity in depletion modeling. After the required temporal fidelity was established, multiple void profiles were used to examine the effect on cask reactivity. The results of these studies are being used to develop recommendations for conservatively modeling the void profile effects for BWR depletion calculations.

The second operational parameter studied was control blade history. Control blades are inserted in various locations and at varying degrees during BWR operation based on the core loading pattern. When present during depletion, control blades harden the neutron spectrum locally because they displace the moderator and absorb thermal neutrons. The investigation of the effect of control blades on post operational cask

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reactivity is documented herein, as is the effect of multiple (continuous and intermittent) exposure periods with control blades inserted. The coupled effects of control blade presence on power density, void profile, or burnup profile will be addressed in future work.

KEYWORDS
BWR burnup credit, storage, transportation, spent nuclear fuel

1. INTRODUCTION

Oak Ridge National Laboratory (ORNL) and the US Nuclear Regulatory Commission (NRC) have initiated a multiyear project to investigate application of burnup credit (BUC) for boiling water 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 is intended to investigate the applicability of peak reactivity methods currently used in SFPs to transportation and storage casks and to validate reactivity calculations and calculated spent fuel compositions within these methods. A paper summarizing the phase 1 peak reactivity investigation is being presented at this conference [1], and a more complete analysis of peak reactivity methods has been published in NUREG/CR-7194 [2]. The second phase focuses on extending BUC beyond peak reactivity. This paper documents analysis of the effects related to moderator density axial profile and to control blade insertion 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 burnup credit" refers to BWR fuel beyond peak reactivity burnup values, typically greater than 15 to 20 GWd/MTU. (The exact burnup point at which a licensee would utilize 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 gadolinium burnable absorber used in BWRs. Due to the burnable absorber, the reactivity of the fuel assembly increases from the beginning of irradiation until the gadolinium has been depleted to a level at which reactivity peaks. After the peak, the reactivity of the fuel decreases for the remainder of operation.

The analyses summarized in this paper focus only on burnup values that occur after the reactivity peak (referred to as "extended" or "full" BUC), which has been used for PWRs for a number of years. In BWRs, the limiting fuel assemblies may be those that initially contain little gadolinium at beginning-of-life, leading to relatively high peak reactivity at a low burnup. The difficulty in applying peak reactivity methods to the limiting fuel assemblies leads to the need for full BUC for BWRs. In order 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 in many ways. This paper analyzes two major operational effects in BWRs: the coolant density axial distribution and the use of control blades during operation. BWR fuel assemblies undergo a significant change in axial moderator density from the bottom to the top due to boiling of the water coolant. Low moderator density leads to increased plutonium production for the same exposure, leading to effects on cask reactivity. Additionally, BWRs operate using control blades (inserted from the bottom of the core) to control excess reactivity. The use of control blades significantly shifts the power distribution in the reactor and also leads to increased plutonium production due to the spectral hardening.

1.1 Code Descriptions

The SCALE code system [3] was used for all calculations performed in these studies. The k_{eff} values in the studies presented in this paper were obtained by performing depletion calculations for each axial segment of the fuel assembly and subsequently transferring the isotopic number densities to three-dimensional (3D) criticality models with the appropriate axial placement of each segment.

The SCALE/TRITON 2D lattice physics sequence was used for depletion calculations in many places. Where possible, SCALE/STARBUCS was used for some depletion calculations (using libraries generated with SCALE/TRITON). SCALE/KENO was used for all reactivity calculations for cask geometries.

The depletion calculations for the axial moderator density study were performed using STARBUCS, which provides an automated linkage between ORIGEN-ARP depletion calculations and KENO criticality safety calculations. STARBUCS creates depleted fuel compositions based on ORIGEN libraries, a fresh fuel description, and an irradiation history. The sequence performs a depletion and decay calculation for each axial node using the ORIGEN-ARP methodology and generates composition input that can be used directly in KENO.

A limitation of STARBUCS is that it can only use a single ORIGEN library in a calculation. Because of its part-length fuel rods, the GE-14 10×10 fuel assembly contains two main axial regions: the “full” or “dominant” region (DOM) and the “vanished” region (VAN). The DOM region, at the bottom of the fuel assembly, contains a full array of fuel pins. The VAN region is located above the level of the part-length fuel rods, so some fuel pins are “vanished” from the lattice. Since the GE-14 assembly design type studied in this work is modeled with two lattices—DOM and VAN—two STARBUCS calculations are necessary to generate depleted compositions for the entire assembly. The compositions resulting from the two calculations are combined to make one set of compositions to represent the depleted and decayed fuel compositions in a single KENO calculation. This same limitation eliminates the ability to use STARBUCS for calculations that require insertion and removal of a control blade, as only a single library can be used. For the control blade analyses, SCALE/TRITON calculations were performed for each axial node in the assembly.

TRITON was used to automate resonance self-shielding, 2D neutron transport, and fuel depletion calculations for each time step of the depletion calculations for the moderator density temporal fidelity study, discussed in Section 2, and for the control blade effects, discussed in Section 4. Many steps are simulated in sequence in order to perform full depletion calculations simulating exposure in the reactor. The version of TRITON in SCALE 6.2 Beta 3 was used for the calculations because it features a new “swap” function. The swap function is a simpler way to define a material exchange in a TRITON timetable block, which is needed for the control blade calculations. After TRITON depletion calculations are completed, a following stand-alone ORIGEN calculation is used to decay fuel compositions for 5 years before the compositions are used in the KENO criticality calculations.

The KENO Monte Carlo code was used to perform k_{eff} eigenvalue calculations of the depleted BWR fuel from TRITON in the GBC-68 cask, which was developed at ORNL as a generic computational benchmark model BWR spent fuel cask [4]. In those calculations, each fuel assembly was modeled with both the DOM and VAN regions, and each of the 68 fuel assemblies in the model was assumed to be identical, which provided conservative estimates of reactivity. All calculations were performed using the SCALE 238-group ENDF/B-VII.0 neutron cross-section library. Actinide-only (AO) and actinide plus fission product (AFP) compositions were used in all studies to determine appropriate modeling approaches for both BUC analysis techniques. The AFP set contains gadolinium, including residual burnable absorber material; the AO set contains no gadolinium. A complete listing of the isotopes in the AO and AFP compositions can be found in Ref. 2.

1.2 Model Descriptions

The GE-14 fuel design type is the only assembly design type modeled in this work. This selection is based on the design’s prevalence in currently operating US BWR plants. While magnitudes of the effects observed in these studies may vary some for other fuel design types, general trends should be consistent.

Thus modeling and design parameters determined to be conservative for the GE-14 assembly design should also be conservative for other fuel design types, although additional calculations may be necessary to confirm this assumption.

The cask used for storage and transportation configuration calculations is the GBC-68 generic BWR fuel cask [4]. In many ways the GBC-68 is analogous to the GBC-32 cask developed for use in PWR BUC analyses [5]. The cask design includes 68 fuel storage cells with a B_4C/Al neutron absorber panel inserted between the faces of adjacent storage cells. The absorber panels and basket are assumed to have the same height as the active fuel stack in the fuel assemblies. The cask body and basket are stainless steel.

1.3 Operating Data

Operating data for a modern BWR were used extensively throughout this research. The data are in the form of nodal core simulator core-follow calculations based on actual plant data that included variables such as moderator density, control history, and power history for every axial node of a fuel assembly (25 total) and for every assembly in the core.

2. MODERATOR DENSITY TEMPORAL FIDELITY STUDY

A temporal fidelity study was performed to determine how frequently the moderator density distribution must be updated during depletion calculations to obtain accurate estimate of k_{eff}. The temporal fidelity study is used to determine the number of steps required to accurately capture the time-varying nature of the moderator density distributions. For this study, the most limiting moderator density profiles are those that are highly variable (i.e., those that change significantly from beginning-of-cycle to end-of-cycle).

To begin this study, we first estimated the fission distribution in a fuel cask filled with depleted BWR fuel whose burnup profile was taken from an actual assembly in the operating data (see Fig. 1). The axial fission distribution results in Fig. 1 were generated using STARBUCS and the GE10x10 ORIGEN library distributed with SCALE. The fuel represented in Fig. 1 is based on the GE-14 design and has been depleted to 35 GWd/MTU. From Fig. 1, it is clear that the majority of fissions in a flooded BWR fuel storage cask are likely to occur near the top of the cask; this effect is known as the "end effect," which is well documented for PWR fuel. The results in Fig. 1 indicate that conditions in the upper portion of the fuel assembly are likely to have a high importance on the reactivity of the fuel cask. The high importance on cask reactivity is primarily due to the burnup profile, which contains relatively low-burnup fuel at the axial ends of the assembly. The fuel at the upper end is more reactive because it contains more plutonium, which is due to the harder neutron spectrum produced by lower density moderator in the top of the core.

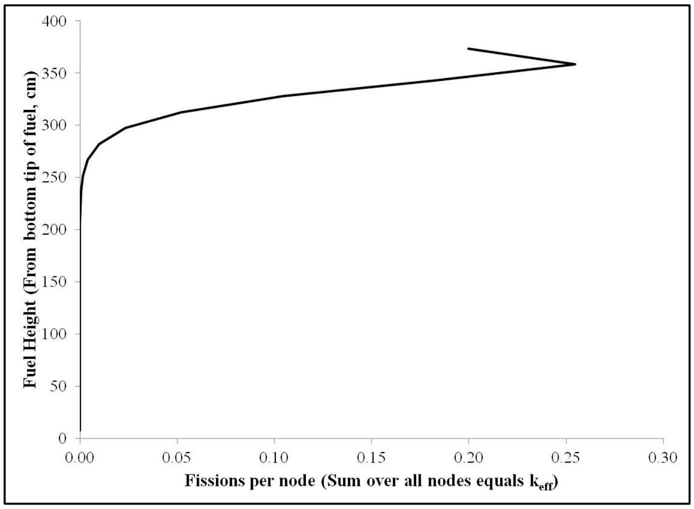

Figure 1. Axial Fission Distribution for BWR SNF in GBC-68.

Because the axial fission distribution in Fig. 1 is higher near the top of the assembly, the k_{eff} calculations used to study the axial void profile are likely to be most heavily influenced by profiles with high void fractions in the top portions of the assembly for an extended period of operation. Analysis of the available axial void profiles have shown that the profiles that have high cycle-average exit void fractions have inherently low variability from beginning-of-cycle to end-of-cycle and are therefore likely to be excluded from the temporal fidelity study despite being of significant interest in conservative cask reactivity determinations.

Because the upper portion of the fuel assembly has a high importance to cask criticality due to the axially peaked fission distribution, axial profiles that have a significant change in the exit void fraction over the cycle were chosen for the temporal fidelity study. A search algorithm was used on the operating data to determine which fuel assemblies underwent the maximum void fraction change over the cycle for the uppermost axial node. In general, the third-cycle fuel assemblies have more significant axial void fraction variability over a single cycle than the first- or second-cycle fuel assemblies, but they also tend to result in lower cycle-average void fractions than the first- or second-cycle fuel assemblies. Due to those complexities, axial void profiles from two different channels were chosen for the temporal fidelity study. The axial void profile from the fuel channel with the highest variability but a relatively low average void fraction, denoted “most variable” (MV), and the axial void profile from a fuel channel with relatively high variability and high average void fraction, denoted “high average” (HA) were chosen for the temporal fidelity study. The two selected axial void fraction profiles can be found in Fig. 2, with the high-exit-variability MV case on the left and the high-integral void fraction HA on the right. Testing profiles that are highly variable over a cycle ensures that the resulting temporal fidelity required for modeling will be sufficient for all other fuel channels that have lower void fraction variability.

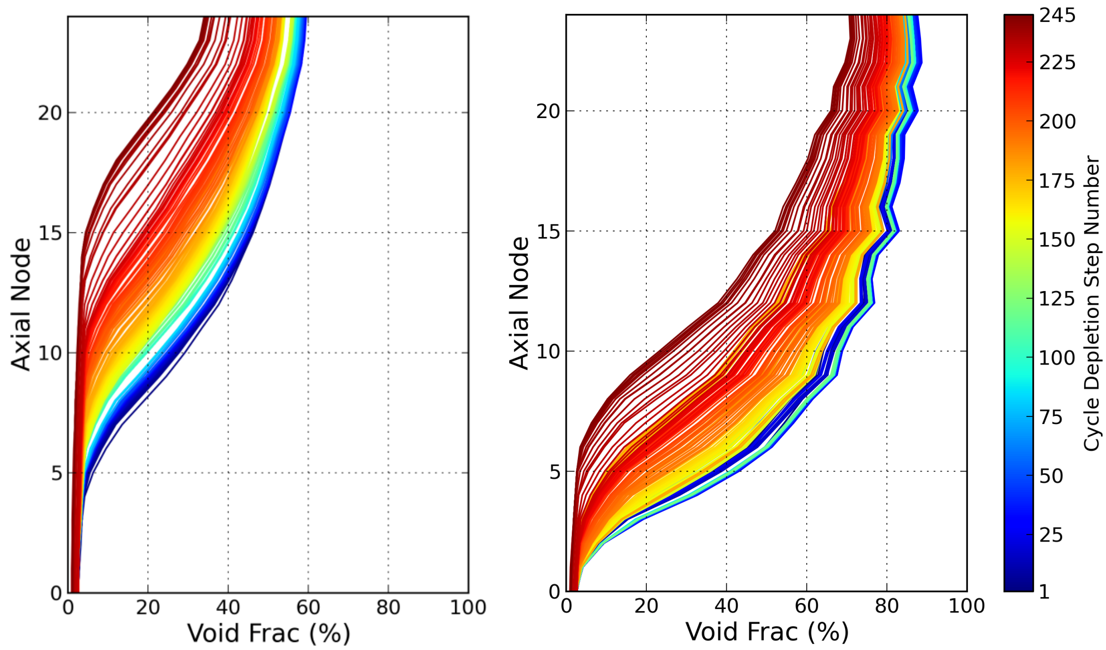

Figure 2. Selected Axial Void Profiles Used for the Temporal Fidelity Study with the Maximum Exit Void Fraction Variance Case (Most Variable) on the Left and the High-Variance and High-Integral Void Fraction Case (High Average) on the Right.

The sensitivity to temporal discretization was examined by performing two sets of calculations with the MV and HA axial void profiles. The first set of calculations used a cycle-average void profile; the second set used a 25-step temporally discretized void profile. Each of these models used 23 depletion steps per cycle and was depleted for three cycles (69 total depletion steps) with a zero-power decay of 30 days between consecutive cycles. Each depletion step was 30 days in length, so a full cycle lasted 690 days and reached a burnup of approximately 15.5 GWd/MTU. The number of depletion steps per cycle (23) is held constant across all TRITON calculations, but the moderator density is temporally discretized into a 1-step (cycle-average) or 25-step scheme. Herein, "25-step" and "cycle-average" refer only to the temporal treatment of the moderator density profile, not the number of depletion steps.

In Fig. 3, moderator density modeling methods (left) and the Pu-239 atom density are plotted for the 25-step and the cycle-average cases for the third axial node from the top of the core in the MV case. That node extends from approximately 320 cm to 335 cm above the bottom of the fuel. Figure 3 shows that the cycle-average void fraction performs quite well when compared to the 25-step reference case, with differences of ± 1% for the target burnup values. The line plotting the difference in Pu-239 between the two cases shows a distinct trend. At the start of each cycle and at the end of each cycle, the Pu-239 atom densities are very close. This occurs because at the beginning and end of each cycle, the integral void history is the same for each case. The three dips occur because the same cycle is being repeated three times. At these points, the area under each of the curves plotted in Fig. 3 is equal. At burnup points that lie between the beginning and end of each cycle, the void history is not equal, leading to a bias. These results suggest that if a true average void fraction can be obtained for each node in the assembly, then the prediction of the Pu-239 isotopics will be accurate. The true measure of whether the cycle-average void fraction approach provides sufficient accuracy will be determined by the fuel cask criticality calculations, which take into account all isotopes rather than just Pu-239. The assessment of Pu-239 is included only to provide some background regarding the physics of the different temporal void fraction modeling approaches.

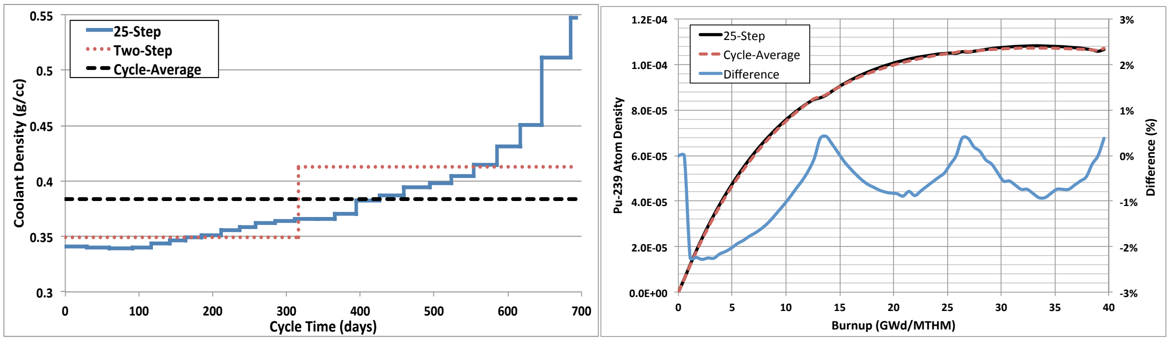

Figure 3. Coolant Density as a Function of Cycle Time (Left) and Pu-239 Atom Density as a Function of Burnup (Right) for the MV Case. The Percentage Difference between the 25-Step and Cycle-Average Cases Is Plotted on the Secondary Axis.

The depleted isotopic number densities that result from the TRITON depletion calculations are used in KENO calculations to determine the k_{eff} value for the GBC-68 cask model loaded with fuel depleted with the different profiles and modeling approaches. The KENO calculations are performed for both the HA and MV profiles using both the detailed 25-step depletion calculations and the cycle-average calculations, for both the AO and actinide and AFP isotope sets (a complete listing of the isotope sets can be found in Ref. 2).

The results for the HA and MV profiles are shown for the AFP isotope set in Fig. 4. The k_{eff} values for the average and detailed depletion calculations are shown with a solid blue line and a dashed red line, respectively. These lines are read on the primary vertical axis and show close agreement throughout the full range of burnups calculated. The difference between the two k_{eff} values, shown with the dotted green line, is read on the secondary vertical axis. The uncertainty in each KENO-calculated k_{eff} is approximately 0.0001 Δk_{eff}, so the uncertainty in the difference is just under 0.00015 Δk_{eff}. Therefore, any difference less than 0.00030 Δk_{eff} represents less than two standard deviations. The difference between the k_{eff} values is small, and the detailed depletion is usually slightly more reactive than the average depletion. The largest difference between the two sets of results is approximately 0.0013 Δk_{eff} for the AFP isotope set at 25.3 GWd/MTU burnup for the HA profile; the detailed moderator density treatment is more reactive. The average moderator density treatment is never more than 0.0005 Δk_{eff} more reactive than the detailed treatment. Similar but smaller differences were calculated for the AO isotope set for both tested axial profiles.

The results of this study demonstrate that burnup-averaged moderator densities can be used in each axial node despite the large variability of axial moderator profiles over a single cycle of operation and between cycles. This approach greatly simplifies depletion calculations. The averaging considered in this study is for each axial node over single cycle; that is, the axial profile is conserved, but the moderator density is averaged in time for each node in the assembly. The moderator profile is maintained axially. A fuel assembly typically experiences several cycles of operation, with moderator density profiles varying within each cycle. The first cycle or two are high-power cycles with a relatively high average void fraction but correspondingly lower variability, similar to the HA profile. The last cycle will see a larger change in moderator densities as the assembly average power plummets near end of life for the assembly, as seen in the MV profile. The depletion calculations performed to examine these profiles effectively assumed that each profile was experienced for three cycles of operation, which is not physically possible. The largest difference resulting from the calculations performed in this study is just over 0.1% Δk_{eff}, indicating a very small bias due to utilization of a cycle-average moderator density in each node. A small penalty applied to k_{eff} should be sufficient to cover any potential non-conservatism resulting from using cycle average moderator density profiles for assemblies that have experienced typical use in a BWR core

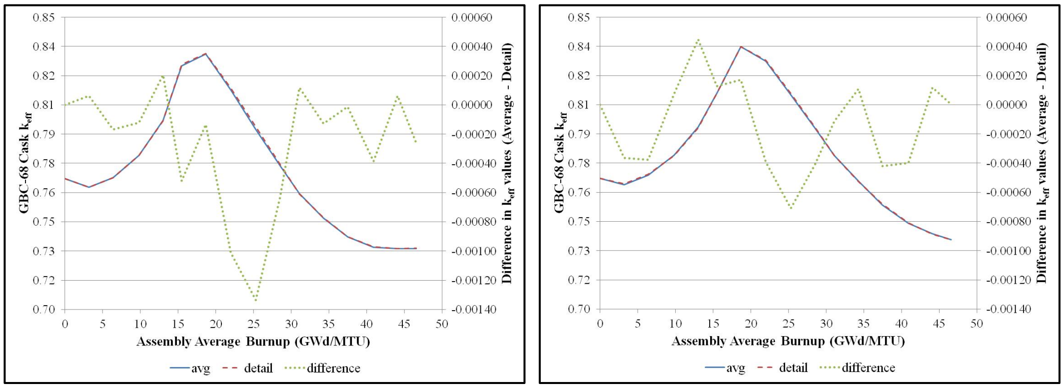

Figure 4. GBC-68 Results for the HA (Left) and MV (Right) Moderator Profiles for the AFP Isotope Set.

3. EFFECT OF MODERATOR DENSITY AXIAL PROFILE

The results of the temporal fidelity study indicate that cycle-average moderator density profiles can be used to make accurate reactivity predictions for discharged BWR assemblies. This conclusion allows the use of STARBUCS to study the effects of a range of moderator profiles because STARBUCS cannot model variable moderator densities as a function of burnup. The STARBUCS calculations are all performed with the same axial burnup profile and specific power to isolate the effect of the axial moderator profile. The specific power and the burnup profile were taken from the assembly that generated the MV moderator profile used in the temporal fidelity study.

As shown in Fig. 1, the upper portion of a discharged BWR assembly has the highest importance to reactivity in a storage cask or transportation package for typical discharge burnups. It has also been shown that low moderator densities lead to higher discharged assembly reactivity [7]. A set of cycle-averaged axial moderator profiles was selected to include both those with low moderator density in the upper portions of the assembly and all profiles that had at least one node with the minimum density for all profiles. Other profiles were added based on an engineering judgment assessment that the additional profiles were potentially limiting. Figure 5 shows the 10 actual assembly profiles that were utilized.

Three additional profiles were created to explore various aspects or approaches to axial moderator distribution modeling. The minimum density profile was created by using the minimum moderator density in each node for all axial profiles. This approach was considered because it represents a simple analysis technique that might be useful, especially for fixed assembly inventories, if the resulting profile conservatively bounds all actual profiles without introducing a large penalty. The second constructed profile was an average profile created by averaging the moderator density in each node across all profiles. This approach was expected to be nonlimiting, but its reactivity effect was quantified to assess its impact. The third constructed profile was a uniform profile equal to a 40% void fraction. It was also expected to be nonlimiting but was investigated because a 40% void fraction is frequently cited as a core average value.

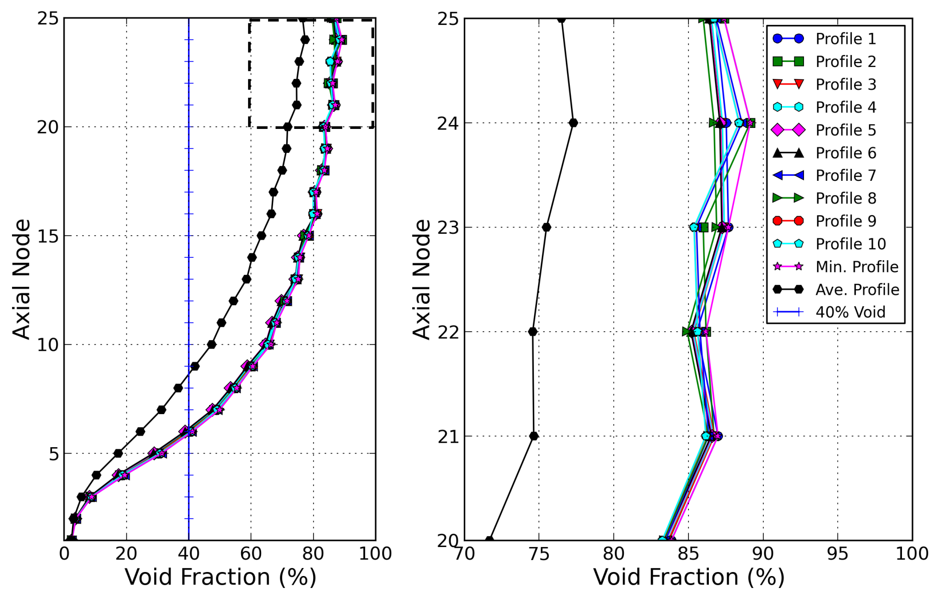

Figure 5. Void Fraction Profiles Used in the Moderator Density Study with the Full Axial Length on the Left and Expanded to Highlight the Upper Portion of the Assembly on the Right.

Results were generated at 30, 40, and 50 GWd/MTU with both the AO and AFP isotope sets. The results for the 10 assembly profiles are provided in Table I for the AFP isotope set. The reactivity rankings of the profiles are very similar, although not identical, across all burnups and with both sets of isotopics, indicating that the effect of the moderator profile is consistent with respect to those variables. The k_{eff} differences are small, as are the differences in the moderator profiles used to generate them. Profile 1 (plotted with blue circles in Figure 5) is the most limiting real profile and is used as the reference for calculating delta-k_{eff} values for all profiles. Although results for the AO isotope set are not shown, they are very similar to the AFP set.

The results indicate that the minimum density profile can be used to conservatively bound the reactivity of the actual assemblies at all burnups and with both isotope sets. The margin provided by the minimum density profile is small for this set of assemblies. The difference from the real profiles for the AFP isotope set is 0.07% Δk_{eff} at 30 GWd/MTU, increasing to 0.14% Δk_{eff} at 50 GWd/MTU. The minimum density profile yielding a more reactive assembly is expected and is a potential approach to simplifying the treatment of axial moderator profiles in BWR BUC, especially considering the small penalty associated with the approach. Unfortunately, the small margin provided does not provide a significant hedge against more limiting profiles or a large margin to offset potential uncertainties in predicted moderator density distributions. Nevertheless, the minimum density profile approach is viable. The primary challenge for its application by a cask vendor is likely to be the collection of a significant and appropriate database of moderator density profiles.

The average void profile is clearly nonconservative, as expected. The average moderator density profile ranges from 2.0% to 2.9% Δk_{eff} less reactive than Profile 1 for the AFP isotope set. This is evidence that discharged assembly reactivity is highly sensitive to moderator density profile and that lower density leads to higher cask reactivity when the same axial burnup profile is considered. In the top node, the average profile has approximately 67% higher moderator density than the minimum case (Profile 2) and nearly 62% higher moderator density than the limiting profile (Profile 1). The absolute differences are on the order of 0.08 g/cm^3, but at the high exit void fractions that difference represents a large relative change. Averaging of moderator densities from different profiles is clearly not an acceptable approach to treating the axial moderator density in BWR BUC.

A constant density representing a 40% void fraction leads to even larger underpredictions of discharged assembly reactivity. For the AO isotope set, the difference from the limiting actual profile is approximately 6.9% Δk_{eff} at 30 GWd/MTU; it increases to almost 10% Δk_{eff} at 50 GWd/MTU. These differences are somewhat smaller in the AFP case, ranging from 4.8% to 7.4% Δk_{eff}. This core average 40% void fraction yields a higher moderator density in the top three nodes of the core than any actual assembly profile in the utilized operating data. The calculated reactivity range for all explicit moderator profiles is likely slightly less than the difference shown between this average value and the limiting case. This case provides additional proof that increasing moderator densities via any averaging approach is likely to yield nonconservative k_{eff} calculations for discharged fuel assemblies.

Table I. GBC-68 cask k_{eff} values for 13 assembly profiles for AFP isotope set with delta-k_{eff} from Profile 1

Profile Name	30 GWd/MTU		40 GWd/MTU		50 GWd/MTU	
	k_{eff}*	Δ-k_{eff}**	k_{eff}	Δ-k_{eff}	k_{eff}	Δ-k_{eff}
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

*All KENO standard deviations less than or equal to 0.0001.
**All delta-k values calculated using Profile 1 as the reference.

4. EFFECT OF CONTROL BLADE USE

Due to a limitation in STARCBUCS (i.e., an inability to change libraries as a function of depletion), SCALE/TRITON must be used to perform depletion calculations for the analysis of control blade use. This is accomplished using the "swap" function that is available in the upcoming SCALE 6.2 release. Thus far, various simplistic control blade histories have been analyzed. The control blade history in the context of this research is defined by the control blade insertion elevation as a function of time. In order to model this in TRITON, input files for each axial node are generated, and then a timetable that corresponds to the insertion depth is applied. For example, for a control blade that is inserted halfway for the entire lifetime, all of the nodes (input file) that correspond to the bottom half of the assembly would be rodded for the entire calculation and all the nodes that correspond to the top half of the assembly would be unrodded for the entire calculation.

Thus far, a set of 22 constructed control blade histories have been simulated. A description of the control blade histories can be found in Table II. In Table II, the insertion elevation is the number of nodes into which the control rods have been inserted from the bottom of the assembly (1 is the bottom node, 25 is the top node). Because ¼-, ½-, and ¾-insertion elevations occur in the middle of certain nodes, it has been assumed that the rod is inserted fully in that particular node, leading to nodes 7, 13, and 19, which roughly approximate ¼, ½, and ¾ insertions. The default for the insertion time in Table II is "not inserted," so, for example, "first half" means that the control blade is inserted for the first half of irradiation and removed for the second half, and "13 nodes, out/in every 240 days" means that the control blade is inserted to an elevation of 13 nodes every 240 days, then removed completely for 240 days.

The control blade histories were chosen in order to ascertain the importance of the effects of insertion elevation, length of insertion, and late versus early control blade insertion. For each of the control blade histories, the same axial power profile and depletion step length was used, so each calculation has the same total burnup. Only the control blade insertion was modified for each of these cases.

Table II. Simulated control blade histories

Name	Insertion Elevation	Insertion Times	Name	Insertion Elevation	Insertion Times
History 1	0 nodes (fully out)	entire life	**History 12**	25 nodes	first half
History 2	25 nodes (fully in)	entire life	**History 13**	25 nodes	last half
History 3	7 nodes	entire life	**History 14**	25 nodes	first third
History 4	13 nodes	entire life	**History 15**	25 nodes	middle third
History 5	19 nodes	entire life	**History 16**	25 nodes	last third
History 6	7 nodes	first half	**History 17**	25 nodes	out/in every 360 days
History 7	7 nodes	last half	**History 18**	25 nodes	out/in every 240 days
History 8	13 nodes	first half	**History 19**	25 nodes	out/in every 120 days
History 9	13 nodes	last half	**History 20**	13 nodes	out/in every 360 days
History 10	19 nodes	first half	**History 21**	13 nodes	out/in every 240 days
History 11	19 nodes	last half	**History 22**	13 nodes	out/in every 120 days

The control blade insertion results for the 22 control blade histories can be found in Table III for the AFP and AO isotope sets. In Table III, the results have been sorted from most limiting to least limiting. As expected, the most limiting cases for both isotope sets are when the rods are fully inserted for the entire irradiation time. This is expected because the hardening of neutron spectrum due to inserted control blades leads to increased Pu-239 production. All results are bounded by the fully inserted control blade case (History 2) by greater than 1.1% Δk_{eff} for the AFP isotope set and greater than 1.2% Δk_{eff} for the AO isotope set, indicating that increased control rod insertion over irradiation leads to higher cask reactivity. It can also be concluded that the depth of control blade insertion has a significant impact. A comparison of History 2 (fully inserted) with History 4 (half-inserted) shows that History 2 is more limiting by 2.8% and 3.7% Δk_{eff} for AFP and AO, respectively. It is also clear that late control blade insertion is also limiting. History 13 (fully inserted last half) is more limiting than History 12 (fully inserted first half), as is History 16 (fully inserted last third) compared with Histories 15 and 14 (fully inserted middle and first third). This is due to increased plutonium production near the end of life, without sufficient unrodded operation to deplete the plutonium before discharge.

Table III. Control blade history cask k_{eff} results

AFP		AO	
History Name	**k_{eff}^***	**History Name**	**k_{eff}^***
History 2	0.76466	History 2	0.85947
History 13	0.75324	History 13	0.84714
History 16	0.74763	History 17	0.84157
History 17	0.74685	History 16	0.84097
History 18	0.74632	History 18	0.83966
History 19	0.74578	History 19	0.83956
History 5	0.74345	History 12	0.83473
History 12	0.74013	History 15	0.83257
History 15	0.73943	History 14	0.82998
History 14	0.73641	History 5	0.82996
History 4	0.73637	History 10	0.82214
History 11	0.73139	History 11	0.82203
History 10	0.73133	History 3	0.82189
History 22	0.73122	History 22	0.82187
History 21	0.73117	History 4	0.82187
History 9	0.73116	History 21	0.82186
History 20	0.73116	History 20	0.82186
History 7	0.73111	History 8	0.82186
History 3	0.73110	History 6	0.82180
History 1	0.73109	History 9	0.82174
History 8	0.73107	History 1	0.82174
History 6	0.73106	History 7	0.82167

*All KENO standard deviations less than or equal to 0.0001.

Given these results, it may be difficult to develop a bounding approach other than assuming that the assembly is rodded the entire lifetime. It may be impossible to obtain control blade operating data for every fuel assembly in order to generate a bounding approach that would be appropriate for all fuel assemblies; however, it is fairly unlikely that an assembly will undergo operation with a control blade deeply inserted for the entire life of the assembly. Simulations using actual control blade histories are ongoing, which may provide additional insight to how limiting the fully inserted case is relative to true operation. After these simulations are complete, a more definitive conclusion may be possible.

5. CONCLUSIONS

Section 2 and 3 summarized two separate but related studies: a temporal fidelity study and an axial moderator density study. The temporal fidelity study was performed to determine the effect of utilization of a cycle-average moderator profile in place of a temporally updated moderator density profile for depletion calculations. The results of the temporal fidelity study indicate that cycle-average moderator density profiles can be used in depletion calculations with the addition of a modest reactivity penalty. The axial moderator profile study established the effect of the profiles. The axial moderator density profile study results 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.

In the control blade use study, a number of different constructed control blade histories were tested. The results show that increased control blade history, especially for nodes near the top of the assembly, lead to increased cask reactivity. In addition, control blade insertion late in life is more limiting than early control blade insertion. Control blades fully inserted for the entire lifetime is the most limiting case and is far more limiting than other constructed control blade histories. Comparison to actual control blade histories remains to be completed.

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