Enclosure 2

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Supplement to Comanche Peak Units 1 and 2 Spent Fuel Pool
Criticality Safety Analysis

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Supplement to Comanche Peak Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis

Westinghouse

WCAP-16827-NP, Addendum 1

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Contributors:

Michael G. Anness
J. W. Bocock
V. N. Kucukboyaci

Approved:

Elliott W. Jackson, Manager
Nuclear Design A

Electronically approved documents are authenticated in the Electronic Document Management System

Westinghouse Electric Company LLC
Nuclear Fuel
4350 Northern Pike
Monroeville, PA 15146

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1.0 Objective

This report presents supplementary results to the criticality safety analysis for Region II of the Comanche Peak Units 1 and 2 spent fuel pool racks. The original analysis, documented in Reference 1, utilizes reactivity credit for burnup, rod cluster control assemblies (RCCAs), RackSaver inserts, axial blankets, soluble boron and ^{241}Pu decay. This supplementary analysis will further support the criticality safety of the Comanche Peak Units 1 and 2 Region II spent fuel racks by explicitly quantifying reactivity effects associated with reactivity credits for burnup, RCCAs, axial blankets, soluble boron and ^{241}Pu decay. As RackSaver insert credit is not subject to upcoming licensing efforts, it is excluded from this analysis. The primary objectives of this calculation are outlined below.

- Quantify the reactivity uncertainty due to PHOENIX-P depletion predictions
- Quantify the reactivity margin associated with axial burnup profiles utilized in depletion calculations
- Quantify the reactivity effects from the presence of grids in fuel assemblies
- Quantify the reactivity effects from the presence of wrapper material in storage racks
- Quantify the reactivity effect of considering eccentric fuel assembly positioning in combination with an increased rack cell inner dimension
- Quantify the reactivity effect of representing axially-blanketed fuel assemblies with unblanketed uniform burnup profiles
- Determination of reactivity for postulated accident scenarios with soluble boron present
- Consideration of postulated reactivity scenarios in the oversize inspection cell
- Justify the conservatism of fuel pellet density
- Justify the axial nodalization of depleted fuel representations
- Justify the core operating conditions utilized in depletion calculations
- Justify the 5% burnup uncertainty
- Clarify various discussions present in the original analysis

This supplementary analysis utilizes neutronic models identical to that of Reference 1, unless otherwise noted.

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2.0 Methodology

This section discusses the nuclear design software and key methodologies employed in this analysis to further support the safe loading of fresh and depleted fuel assemblies in the Comanche Peak Units 1 and 2 spent fuel pools.

2.1 General Methodology

The methodology utilized in this report is identical to that of the original analysis in Reference 1, with few exceptions. The two exceptions are the use of SCALE5.1 and PARAGON software, unless otherwise noted in the text. Both of these codes are utilized here to ease the computational burden associated with the methods utilized in Reference 1. SCALE5.1 and PARAGON are utilized as direct replacements for SCALE4.4 and PHOENIX-P, respectively, from Reference 1.

2.2 Use of SCALE5.1 for Spent Fuel Pool Reactivity Calculations

Similar to the SCALE4.4 code utilized in Reference 1, SCALE5.1 was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. The SCALE version that is utilized for this analysis is a code system that runs on Linux computers and includes the control module CSAS25.

The SCALE5.1 code is utilized for reactivity determinations in the spent fuel pool environment. Its use is identical to that of SCALE4.4 in Reference 1, and in many cases identical input decks are used (updated to specify the use of NITAWL). However, as the SCALE5.1 source code has been re-written in a more modern FORTRAN language, the computational run time is greatly reduced when running the code on Linux computers.

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]^{a,c} In this supplementary analysis, reactivity differences for SCALE5.1 results are investigated in great detail. Furthermore, the absolute determinations of reactivity presented in Section 3.7 account for the change in mean calculational method bias between the two codes.

2.3 Use of PARAGON for Reactor Depletion Calculations

PARAGON is a two-dimensional multigroup transport theory code to generate nuclear characteristics for PWR lattices for use in global reactor calculations, and has been approved for use as a direct replacement for PHOENIX-P in reactor physics calculations in Reference 3. [

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3.0 Quantification of Reactivity Effects Excluded from the Original Analysis

The purpose of this section is to describe the analytical approach utilized to quantify the reactivity effects associated with the items outlined in Section 1.0 of this report.

3.1 Justification of PHOENIX-P Depletion Predictions

The original analysis (Reference 1) states that "a 5% burnup measurement uncertainty based on the maximum burnup credited for each initial enrichment in a storage configuration was applied to all the depleted fuel assemblies in that configuration. Since the burnup measurement uncertainty is dependent on the magnitude of the burnup credited in the analysis, it is determined iteratively at each initial enrichment considered in a storage configuration." The burnup measurement uncertainty is then applied with other uncertainties at no soluble boron conditions. Additionally, this uncertainty, along with a fuel depletion uncertainty, is further applied in the determination of soluble boron credit.

The NRC guidance provided in Reference 2 states that "a reactivity uncertainty due to uncertainty in fuel depletion should be developed." Additionally, the guidance indicates that "in the absence of any other determination...an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." To ensure that this reactivity uncertainty is properly accounted for in the summation of biases and uncertainties, it is investigated here.

Table 3-1 presents a comparison of PHOENIX-P isotopic predictions relative to Yankee Core 5 measurements. Since PHOENIX-P's ability to predict isotopics must be converted into terms of reactivity (specifically, Δk_{eff}) for the purpose of a reactivity uncertainty, these differences are applied to the isotopic predictions in the original Comanche Peak Units 1 & 2 criticality safety analysis (Reference 1). To determine the magnitude of the reactivity uncertainty, perturbation calculations are performed relative to the nominal isotopic predictions in the "4-out-of-4" storage configuration. As storage configurations with fewer assemblies or reactivity-suppressing materials will exhibit a dampening effect on any reactivity increases due to the perturbations of isotopic compositions, this storage configuration is chosen to bound reactivity effects for all configurations.

[Table 3-1

Figure 3-1

Figure 3-4]^{a,c}

The perturbation calculations show the absolute k_{eff} value for the perturbed and nominal cases, along with the relative Δk_{eff} difference between the cases. The differences presented in Figure 3-1 through Figure 3-4 are much smaller than the burnup uncertainty values considered in the original Comanche Peak Units 1 & 2 criticality safety analysis (Reference 1). Therefore, it is

determined that the burnup uncertainty in Reference 1 is sufficient to bound the "reactivity uncertainty due to uncertainty in fuel depletion" that is described in Reference 2. The depletion uncertainty considered in the determination of soluble boron requirements – which equates to 0.010 Δk_{eff} units per 30,000 MWd/MTU of burnup – is also larger than the differences determined here, and is therefore also sufficient to account for the reactivity uncertainty described in Reference 2. Furthermore, the reactivity differences presented in Figure 3-4 are sufficiently small such that positive margin is present in the Reference 1 analysis.

3.2 Justification of Axial Burnup Profiles Utilized in Depletion Calculations

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]^{a,c} This axial burnup profile, along with a uniform burnup profile, is then utilized in all depletion calculations. The largest reactivity between the profiles is then chosen for consideration in determining burnup requirements for safe storage of depleted fuel.

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To properly justify the conservatism of the axial burnup profile [ ]^{a,c} considered in the original (Reference 1) analysis, a thorough analysis of axial burnup profiles from the database of Reference 8 is conducted. The database of Reference 8 contains thousands of axial burnup profiles from several reactors, and reactor types, around the world. Since the lattice design, and the reactor type in which it is irradiated, influences the axial burnup profile of fuel assemblies, only axial burnup profiles from Westinghouse 17x17 fuel assemblies, identical to that utilized at Comanche Peak Units 1 and 2, are considered in this investigation. Furthermore, only the limiting axial burnup profiles from this assembly design are considered. The limiting axial burnup profile is chosen based on the relative burnup of the top two nodes. Fuel assemblies from the database are audited, and the assembly with the minimum relative burnup in the top two nodes is chosen to represent the limiting axial burnup profile for a given burnup range.

The limiting axial burnup profiles for the Westinghouse 17x17 fuel lattice from Reference 8 are then considered in depletion calculations at conditions representative of the uprated Comanche Peak Units 1 and 2 reactor cores. The reactivity effects due to axial burnup profile for burnups less than 46 GWd/MTU are only applicable at low enrichments, since higher enrichments require greater than 46 GWd/MTU of burnup for acceptable storage. Therefore, fuel assemblies of 2.4 w/o ^{235}U enrichment are investigated to determine the reactivity effects of a limiting burnup profile. The limiting axial burnup profiles with associated depletion conditions are summarized in Table 3-2 for each applicable burnup range.

The limiting fuel representations described above are simulated at 2.4 w/o ^{235}U in the "4-out-of-4" storage configuration model from Reference 1. As storage configurations with fewer assemblies or reactivity-suppressing materials will exhibit a dampening effect on any reactivity differences due to axial burnup profiles, this storage configuration is chosen to bound reactivity effects for all configurations. This enrichment is chosen due to its past use in Comanche Peak Units 1 and 2.

Figure 3-5 shows the reactivity of the most limiting axial burnup profiles and that of Profile 1 utilized in Reference 1. Figure 3-5 demonstrates that reactivity can increase due to the consideration of limiting axial burnup profiles. The maximum reactivity increase observed is $0.00928 \pm 0.00048\ \Delta k_{eff}$ relative to the Profile 1 shape used in Reference 1.

The actual axial burnup profile for assemblies stored in the Comanche Peak spent fuel pools is uncertain. The storage of fuel assemblies with varying burnup profiles reduces the resulting reactivity impact of severe burnup profiles due to a misalignment of reactive zones in adjacent assemblies. Due to the uncertainty in burnup shape profile, the reactivity increase of 0.00928 Δk_{eff} is statistically combined with other uncertainties in the analysis.

The statistical combination of burnup profile uncertainty with other uncertainties in the "4-out-of-4" storage configuration at 2.0 w/o ^{235}U, will lead to an increase in total biases and uncertainties of 0.00279 Δk_{eff}.

3.3 Presence of Assembly Grids

The original analysis (Reference 1) states that "no credit is taken for spacer grids". While neglecting the inherent neutron absorption of such structural material typically leads to conservative determinations of reactivity, no quantification is provided for the reactivity effects associated with modification of the H/U ratio in the assemblies or displacement of soluble boron absorber credited in the original analysis.

As fuel assembly reactivity response will vary with neutron spectrum effects due to depletion, proper quantification of these reactivity effects requires the consideration of fuel ranging from low enrichment and zero burnup to 5.0 w/o ^{235}U and high burnup values. As such, this analysis utilizes unirradiated 1.02 w/o ^{235}U fuel and 5.0 w/o ^{235}U fuel at 75.0 GWd/MTU to ensure that enrichment/burnup combinations allowable for storage in the original analysis are considered.

Each fuel representation described above is simulated, with and without grids present, in the "4-out-of-4" storage configuration model from Reference 1. As storage configurations with fewer assemblies or reactivity-suppressing materials will exhibit a dampening effect on any reactivity increases due to the presence of grids, this storage configuration is chosen to bound reactivity effects for all configurations.

The grids are modeled with the least absorptive grid material in use, Zircaloy, smeared with water across a cuboid the approximate size of physical grids. The number of Zircaloy atoms from an explicit representation of a grid is conserved in the smeared representation. Eight grids are modeled in this manner, with equidistant spacing of 18.0 inches across the height of the active fuel length. This fuel representation is shown in Figure 3-6.

The reactivity effects from the presence of grids are determined with the models described above at soluble boron concentrations ranging from 0 to 2000 ppm in increments of 100 ppm. The results of these calculations are shown in Figure 3-7.

The results demonstrate that unirradiated and depleted fuel assemblies are conservatively represented without grids present at no soluble boron conditions. Furthermore, 5.0 w/o ^{235}U fuel at 75.0 GWd/MTU is either more reactive with grids excluded, or any reactivity increase at high soluble boron concentrations is statistically insignificant. Conversely, unirradiated 1.02 w/o ^{235}U fuel is shown to produce reactivity increases when grids and soluble boron are present – the maximum difference is 0.00162 $\pm$ 0.00013 Δk_{eff}. Since nearly all fuel assemblies stored in the Comanche Peak spent fuel pools are of higher enrichments and depleted, this will have minimal impact on the criticality safety margins.

3.4 Presence of Fuel Storage Rack Wrappers

The original analysis (Reference 1) states that "the stainless steel wrappers that are present in the Comanche Peak Unit 2 Region II storage racks are not modeled", but no explicit quantification is provided for the reactivity effects associated with displacement of the soluble boron absorber that is credited.

Similar to the analysis of assembly grids, this analysis utilizes unirradiated 1.02 w/o ^{235}U fuel and 5.0 w/o ^{235}U fuel at 75.0 GWd/MTU to ensure that enrichment/burnup combinations allowable for storage in the original analysis are considered. Additionally, the "4-out-of-4" storage configuration model from Reference 1 is chosen to bound reactivity effects for all configurations.

The wrappers are modeled explicitly with Type 304 stainless steel, and it is filled with water. The wrapper is 0.02 inches thick and 7.53 inches in width. The wrapper representation is shown in Figure 3-8.

The reactivity effects from the presence of wrappers are determined with the models described above at soluble boron concentrations ranging from 0 to 2000 ppm in increments of 100 ppm. The results of these calculations are shown in Figure 3-9.

The results demonstrate that unirradiated and depleted fuel assemblies are conservatively represented without wrappers present.

3.5 Assembly Eccentric Positioning in Combination with Storage Can Tolerances

The original analysis (Reference 1) considers the eccentric positioning of fuel assemblies in storage cells as a reactivity uncertainty case in the determination of biases and uncertainties for each storage configuration. However, the eccentric positioning of fuel assemblies was considered independent of the storage cell inner dimension. To ensure that the reactivity uncertainty associated with eccentric position of fuel assemblies is conservatively accounted for, its effect is investigated here in combination with variations to the storage cell inner dimension.

The reactivity perturbation associated with eccentric positioning was determined to be negative in the original analysis through direct simulation for all storage configurations except the "2-out-

of-4". In this investigation, the eccentric position of fuel assemblies is considered while also varying the storage cell inner dimension to the inner and outer bounds of the physical tolerance. While it is determined that many storage configurations still exhibit a negative reactivity difference, the "2-out-of-4" storage configuration does realize a larger change in reactivity when eccentric positioning is considered in combination with storage cell inner dimensions.

The increase in reactivity associated with these physical tolerances being considered together is 0.00711 Δk_{eff} for the "2-out-of-4" storage configuration. When the reactivity increase associated with eccentric positioning and storage cell inner dimension (considered independently) are replaced with this change in the summation of biases and uncertainties, the total bias and uncertainties value increases by 0.00047 Δk_{eff} relative to that reported in Reference 1.

3.6 Uniform Burnup Representation for Blanketed Fuel Assemblies

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To investigate the reactivity effects of axial blankets in the Comanche Peak Units 1 and 2 spent fuel pool, axial burnup profiles from reactor physics simulations are reviewed. These axial burnup profiles are created with core simulator nodal codes and are presented in Figure 3-10 through Figure 3-13 for Unit 2 and Figure 3-14 through Figure 3-18 for Unit 1. Upon review of the data, the most limiting axial burnup profile is chosen based on the magnitude of the relative burnup in the top two nodes. The most limiting burnup profile is determined to be the center assembly (quarter-core location 1, 1) from Unit 1 Cycle 10.

The limiting fuel representation described above is simulated [ ]^{a,c} in the "4-out-of-4" storage configuration model from Reference 1. A summary of the fuel assembly and depletion characteristics is given in Table 3-3. As storage configurations with fewer assemblies or reactivity-suppressing materials will exhibit a dampening effect on any reactivity differences due to the presence of axial blankets, this storage configuration is chosen to bound reactivity effects for all configurations with axial blankets.

Figure 3-19 shows the reactivity of the most limiting axially-blanketed fuel assembly and that of an [

]^{a,c} The [ ]^{a,c} representation is most conservative at 35 GWd/MTU, reaching a maximum reactivity difference of 1893 $\pm$ 41 pcm Δk_{eff} (1 pcm = 10^{-5}). The least conservative time of life is at 60 GWd/MTU when the reactivity difference is 361 $\pm$ 39 pcm Δk_{eff}.

3.7 Alternative Soluble Boron Credit Formulation

The original analysis (Reference 1) utilizes a soluble boron credit methodology that utilizes differential boron worths to determine the soluble boron requirements to compensate for three required reactivity terms. However, this methodology considers the various soluble boron credit terms as separate reactivity terms and then each is individually translated into corresponding boron concentrations through a "parallel" application. Since the differential boron worth is reduced in the presence of increasing boron concentrations, a more conservative approach is to sum the Δk_{eff} for each of the three terms in a "serial" application, and then translate this integral reactivity worth into an overall soluble boron concentration requirement.

Furthermore, the postulated accident scenarios are considered in the spent fuel pool model with no soluble boron present. This calculational procedure has been shown to produce smaller reactivity increases, in some instances, relative to the consideration of postulated accident scenarios in the presence of a required soluble concentration.

Due to the aforementioned uncertainties regarding the conservatism present in the soluble boron requirement determination of Reference 1, the soluble boron concentration requirements are further investigated here.

The postulated accident scenarios considered in the original analysis include:

- Intra-module water gap reduction due to seismic event,
- Dropped fresh fuel assembly on top of the storage racks,
- Spent fuel pool temperature greater than 150°F including partial voiding,
- Removal of a reactivity suppression device,
- Misloaded fresh fuel assembly into an incorrect storage rack location, or outside the racks.

The intra-module gap reduction need not be explicitly considered here since such an intra-module gap is not considered in the base models. Additionally, a dropped fuel assembly on top of the racks does not require explicit simulation since the racks extend greater than 1 ft. above the top of the fuel assemblies during the stored configuration. This distance is sufficient to isolate the stored fuel assemblies from a dropped fuel assembly – this justification is similar to that of the 1 ft. spacing required between fuel assemblies that is common during fuel movement. While the allowed placement of non-fissile material in empty storage cells may reduce the moderator separating assemblies in this postulated accident scenario, any reactivity increase will not approach that of close-packed, fully-aligned fuel assemblies involved in a misload into an incorrect storage rack location.

The increase in spent fuel pool temperature is explicitly considered and produces decreases in reactivity with decreasing water densities resulting from heatup and boiling. The removal of a reactivity suppression device, such as a RackSaver or RCCA, is also explicitly considered, but the reactivity increase from this accident is approximately 0.01 Δk_{eff}. While this increase is

substantial, it is not nearly as limiting as a fuel assembly misload into an incorrect storage rack location.

Finally, the misloaded fuel assembly of maximum reactivity is considered. Due to the increased neutron leakage in scenarios involving misloaded fuel assemblies outside of the storage racks, the scenario with a fuel assembly misloaded into an incorrect storage location is bounding. This postulated accident scenario produces the largest increases in reactivity, greater than 0.08 Δk_{eff}, and is considered explicitly here in each storage configuration.

This investigation initially considered the differential boron worth for each storage configuration, determined the required soluble boron concentration for a reduction of 0.05 Δk_{eff} units and reactivity uncertainties, and then initiated accident calculations from the determined soluble boron concentration. While this investigation demonstrated that the increase in k_{eff} due to postulated accident scenarios is larger in the "2-out-of-4" and "3-out-of-4" storage configurations than the corresponding scenario at no soluble boron conditions (all others were lower), it was determined that a more pertinent analysis would consider the resulting k_{eff} of each storage configuration when the required concentration of soluble boron is present in the pool water.

The required concentration of soluble boron in the original analysis (Reference 1) is 1607 ppm. When this soluble boron concentration is modeled in each storage configuration, and a misloaded OFA fuel assembly of 5.0 w/o ^{235}U enrichment is considered in an incorrect storage location, the reactivity increase is effectively mitigated. The resulting k_{eff} values for these postulated accident scenarios are shown in Table 3-4. The largest reactivity for configurations other than the "2-out-of-4" when postulated accidents are considered in combination with the required soluble boron concentration from the original analysis is k_{eff} = 0.89421 $\pm$ 0.00025 for the "4-out-of-4-with 2 RCCAs" storage configuration. This value provides at least 0.01 Δk_{eff} of criticality safety margin relative to the k_{eff} requirement accounting for a reduction of 0.05 Δk_{eff} units, the largest reactivity uncertainties and the difference between the bias in SCALE versions (0.945 - 0.03940 - 0.00035 = 0.90525).

The reactivity of the "2-out-of-4" storage configuration when postulated accidents are considered in combination with 1900 ppm soluble boron concentration is k_{eff} = 0.92427 $\pm$ 0.00010. This value results in a reactivity difference of 0.00151 Δk_{eff} relative to the k_{eff} requirement accounting for a reduction of 0.05 Δk_{eff} units, the biases and uncertainties, the 5% burnup measurement uncertainty and the difference between the bias in SCALE versions (0.945 - 0.01962 - 0.00227 - 0.00035 = 0.92276). As this reactivity difference will be offset with existing criticality safety margin, this demonstrates that less than 1900 ppm is required.

3.8 Analysis of Oversize Inspection Cell Postulated Accident Scenario

The original analysis (Reference 1) considers the presence of a fuel assembly in the oversize inspection cell, but does not explicitly consider the postulated accident scenario involving a second assembly dropped into the oversized cell. Since the oversize inspection cell is large enough to accommodate two assemblies in close proximity, the reactivity consequence of this postulated accident is considered here.

Similar to the analysis of other postulated accident scenarios described above, the analysis of postulated accidents in the oversize inspection cell considers an OFA fuel assembly of 5.0 w/o ^{235}U enrichment misloaded into the storage cell. The assembly is modeled adjacent to the existing fuel assembly, and the reactivity with soluble boron present is determined.

The reactivity of this postulated accident scenario is k_{eff} = 0.87958 $\pm$ 0.00019. This value provides at least 0.02 Δk_{eff} of criticality safety margin relative to the k_{eff} requirement accounting for a reduction of 0.05 Δk_{eff} units, reactivity uncertainties and the difference between the bias in SCALE versions (0.945 - 0.03940 - 0.00035 = 0.90525). Note that only 1327 ppm of soluble boron was considered in this determination.

Table 3-1. Comparison of PHOENIX-P Actinide Predictions to Yankee Core 5 Measurements

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Table 3-2. Limiting Unblanketed Axial Burnup Profiles and Depletion Characteristics at Various Burnup Ranges

Zone Number	Height (inches)	Relative Power at Applicable Burnup Range (GWd/MTU)				Moderator Temperature (°F)
		30 - 34	34 - 38	38 - 42	42 - 46	
1	8.0	0.536	0.558	0.699	0.693	559.8
2	8.0	0.895	0.911	0.929	0.955	563.5
3	8.0	1.060	1.066	1.074	1.059	567.1
4	96.0	1.100	1.095	1.074	1.073	590.9
5	8.0	0.988	0.988	1.009	0.987	614.7
6	8.0	0.817	0.824	0.828	0.856	618.3
7	8.0	0.494	0.507	0.557	0.568	622.0

Table 3-3. Limiting Blanketed Assembly and Depletion Characteristics

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Table 3-4. keff Results of Postulated Accident Scenarios in the Presence of the Required Soluble Boron Concentration

Storage Configuration	$k_{eff} \pm \sigma$
"2-out-of-4"	0.92427 ± 0.00010
"3-out-of-4"	0.87234 ± 0.00013
"4-out-of-4"	0.86127 ± 0.00013
"4-out-of-4 with 1 RCCA"	0.87575 ± 0.00025
"4-out-of-4 with 2 RCCAs"	0.89421 ± 0.00025

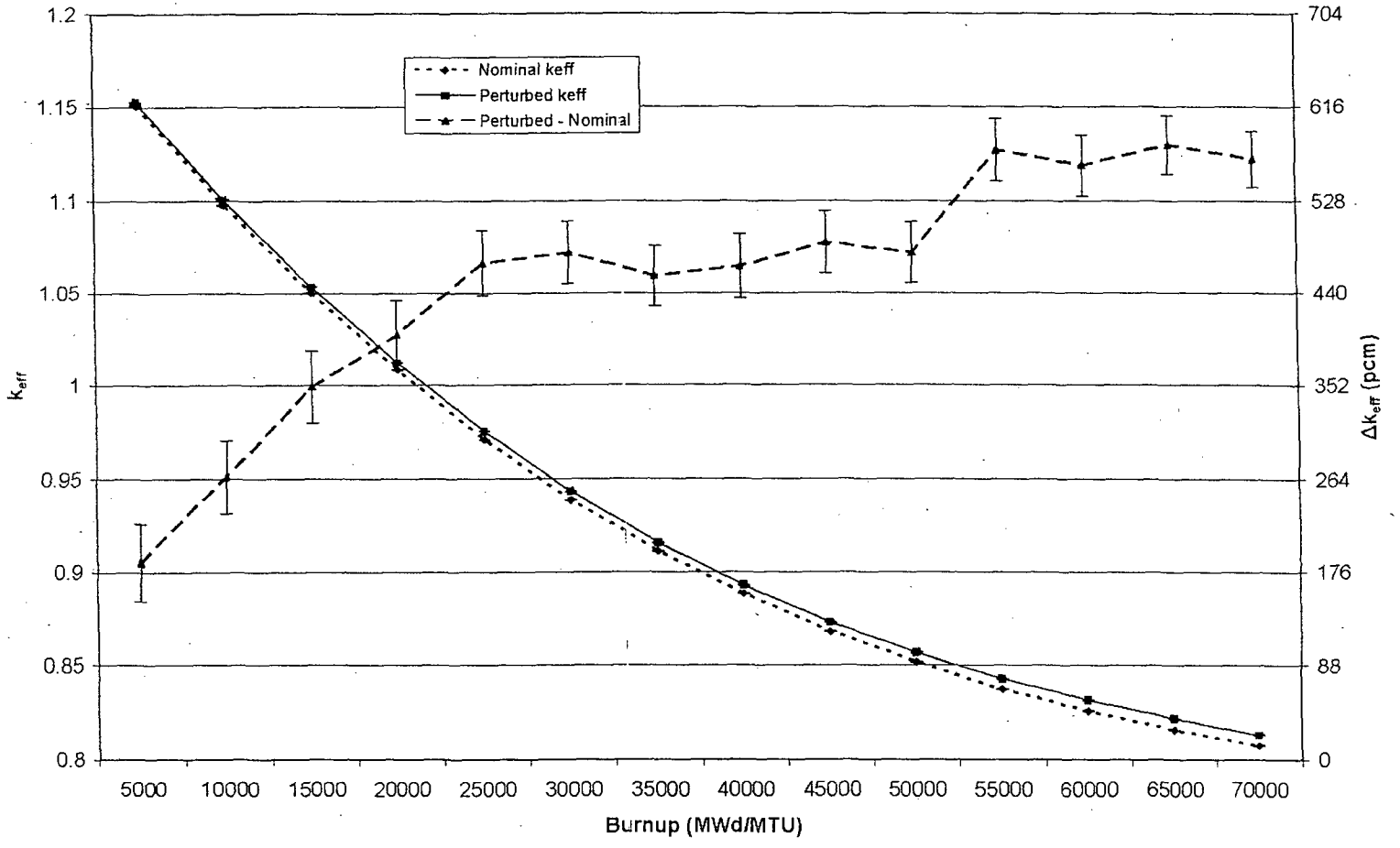

Figure 3-1. k_{eff} Results due to Isotopic Perturbations for 2.0 w/o ^{235}U Fuel in Comanche Peak Spent Fuel Racks

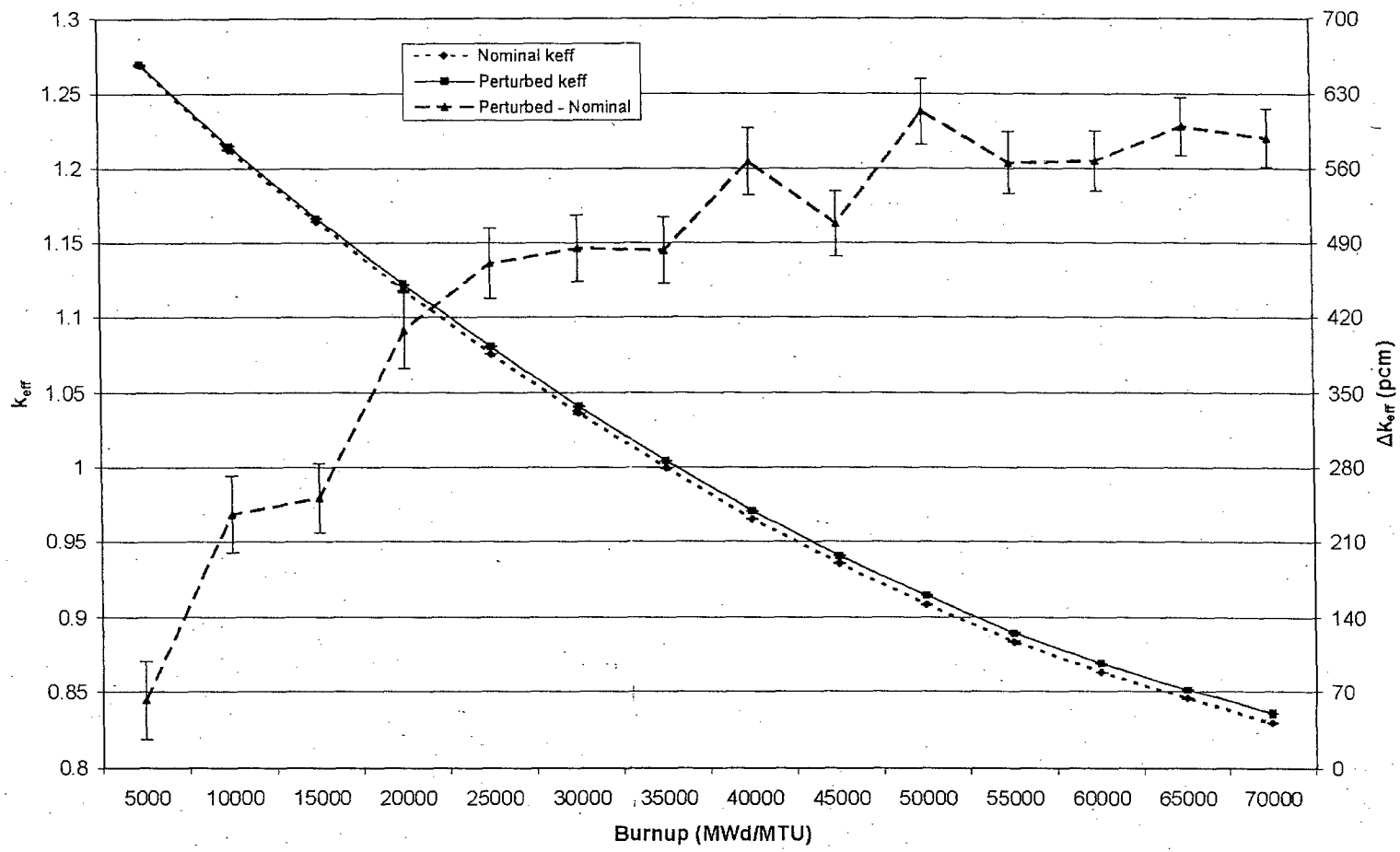

Figure 3-2. k_{eff} Results due to Isotopic Perturbations for 3.0 w/o ^{235}U Fuel in Comanche Peak Spent Fuel Racks

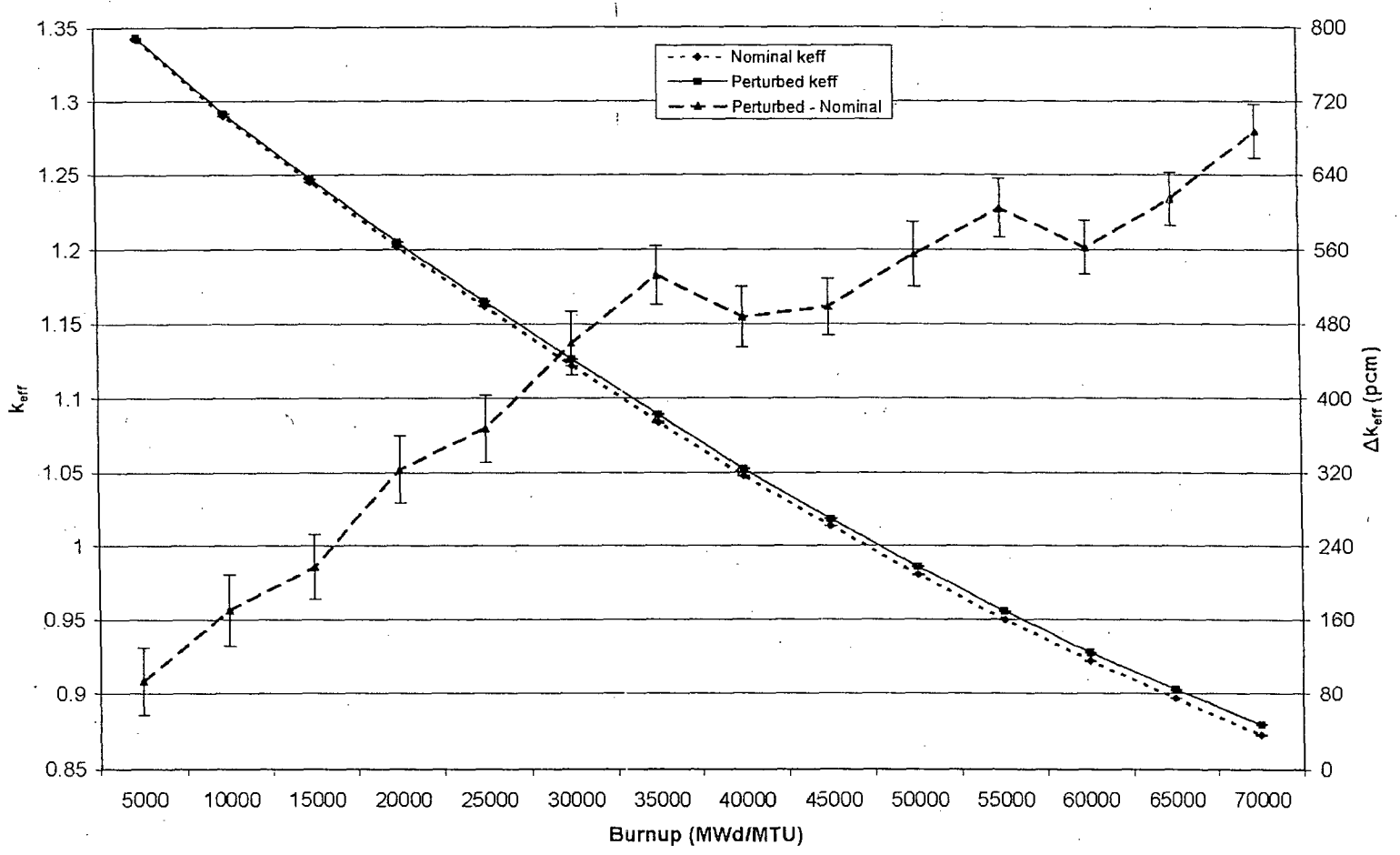

Figure 3-3. k_{eff} Results due to Isotopic Perturbations for 4.0 w/o ^{235}U Fuel in Comanche Peak Spent Fuel Racks

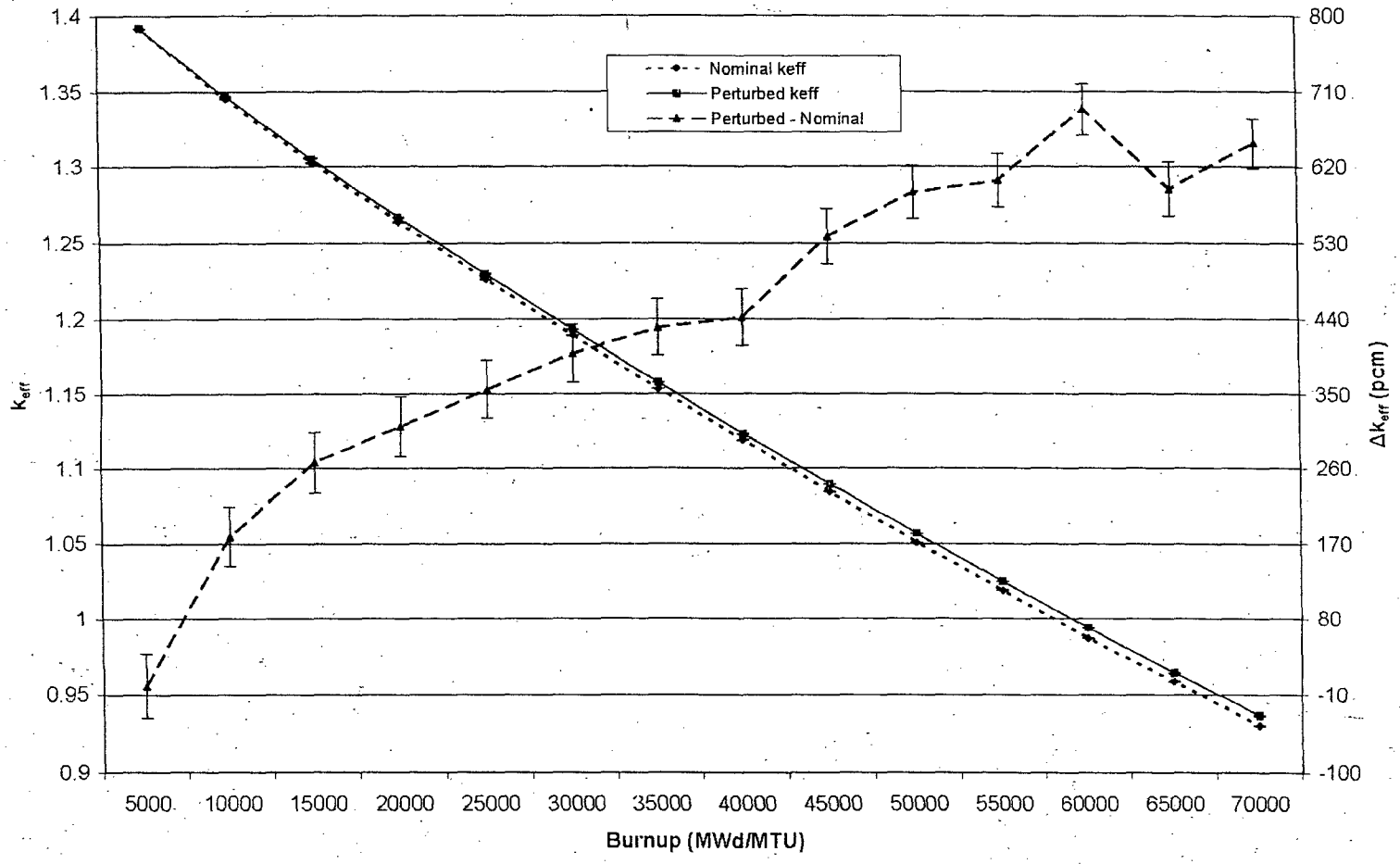

Figure 3-4. k_{eff} Results due to Isotopic Perturbations for 5.0 w/o ^{235}U Fuel in Comanche Peak Spent Fuel Racks

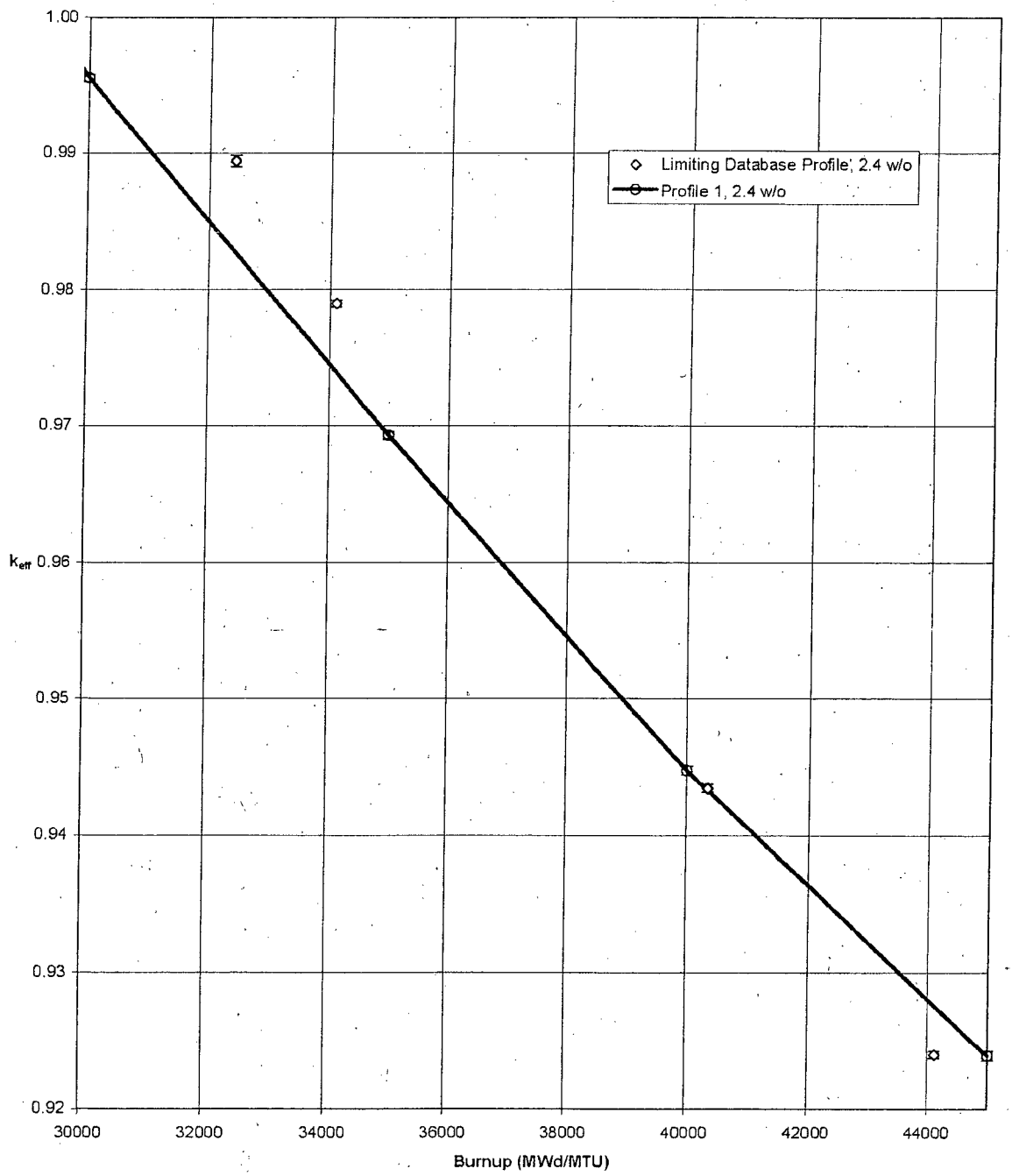

Figure 3-5. Reactivity Comparison of Limiting, Unblanketed 17x17 Westinghouse Axial Burnup Profiles vs. Profile 1 Fuel Assembly Representation

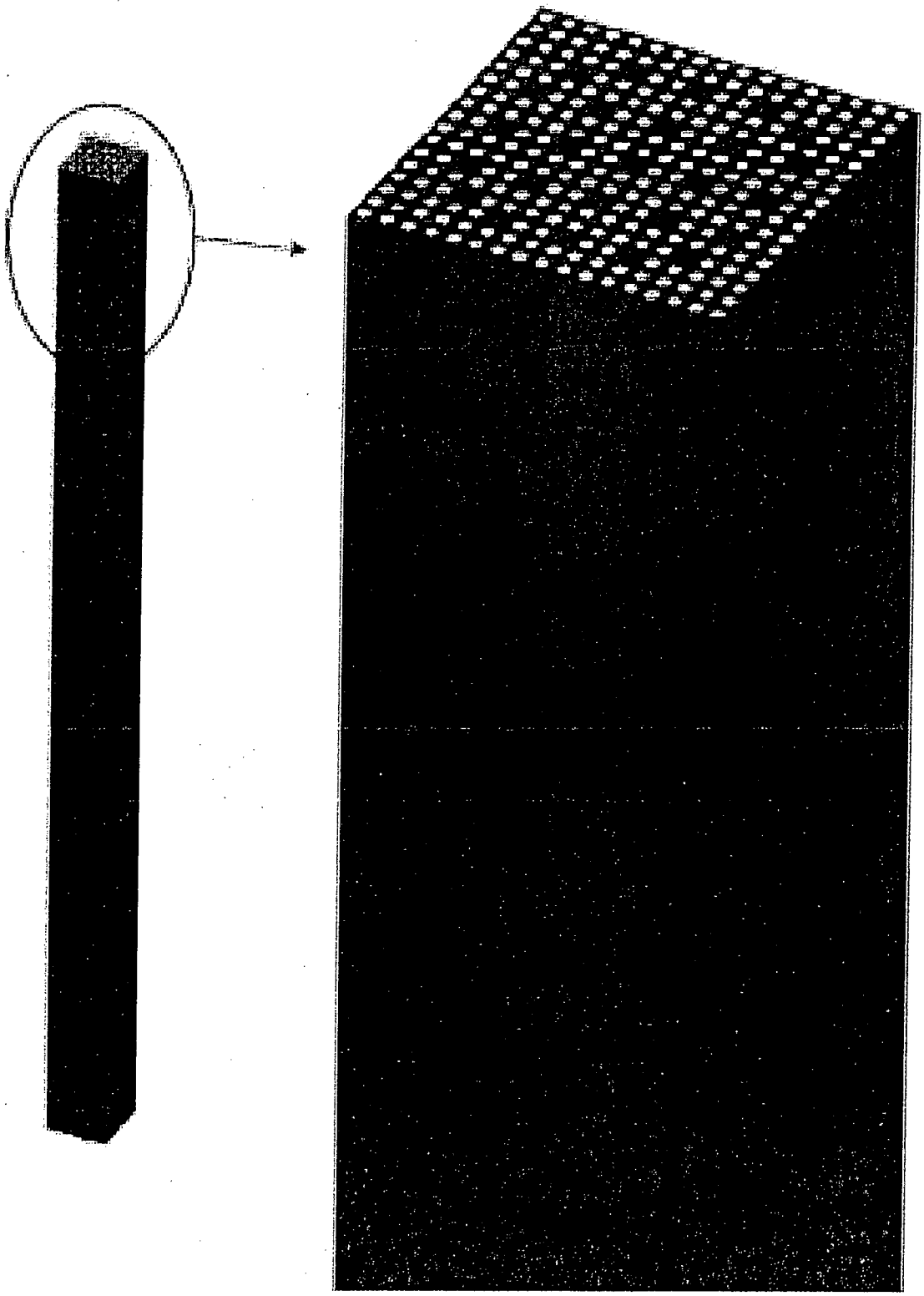

Figure 3-6. Illustration of Assembly Representation with Grids Present

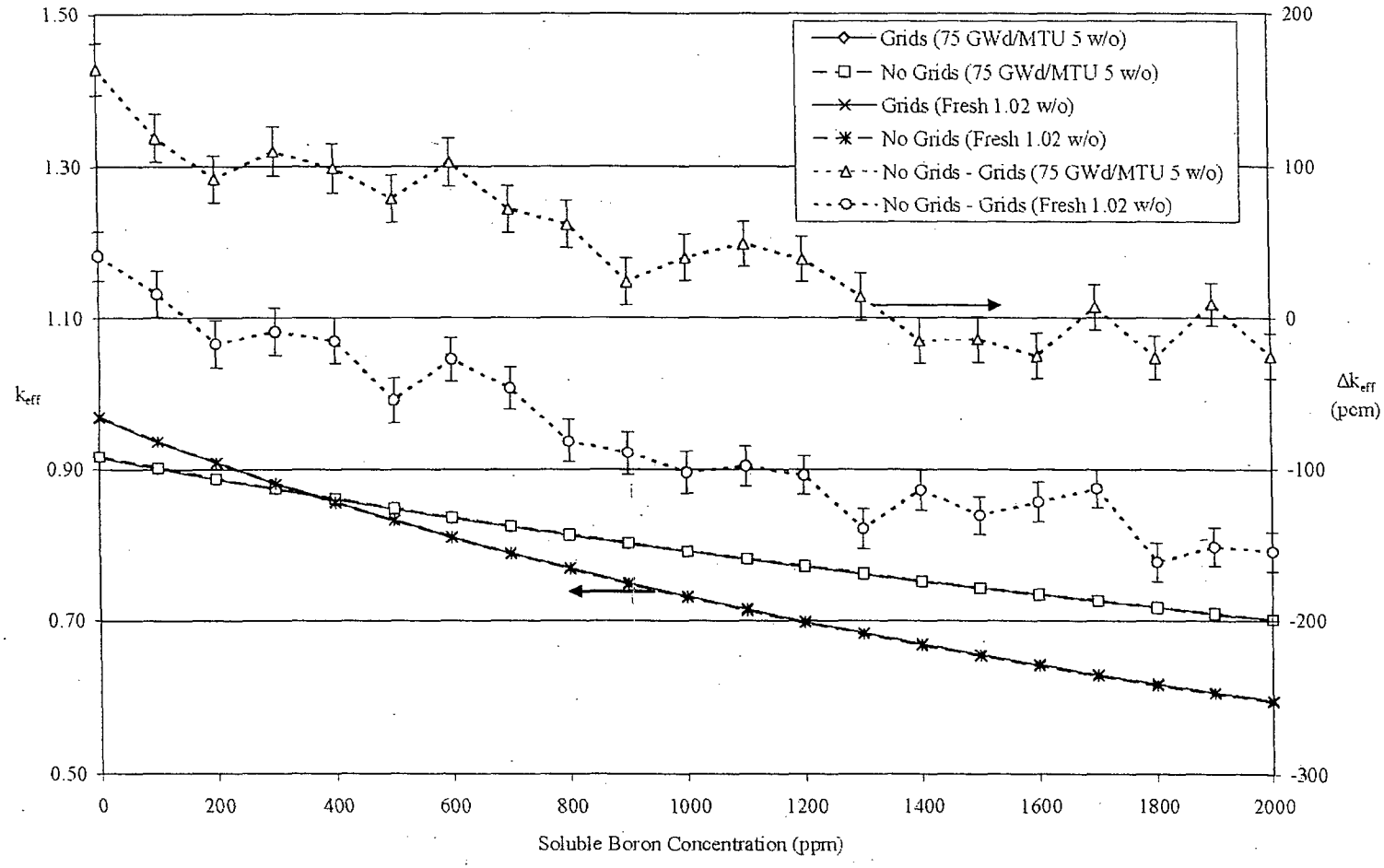

Figure 3-7. Reactivity Effects from Assembly Grids vs. Soluble Boron Concentration

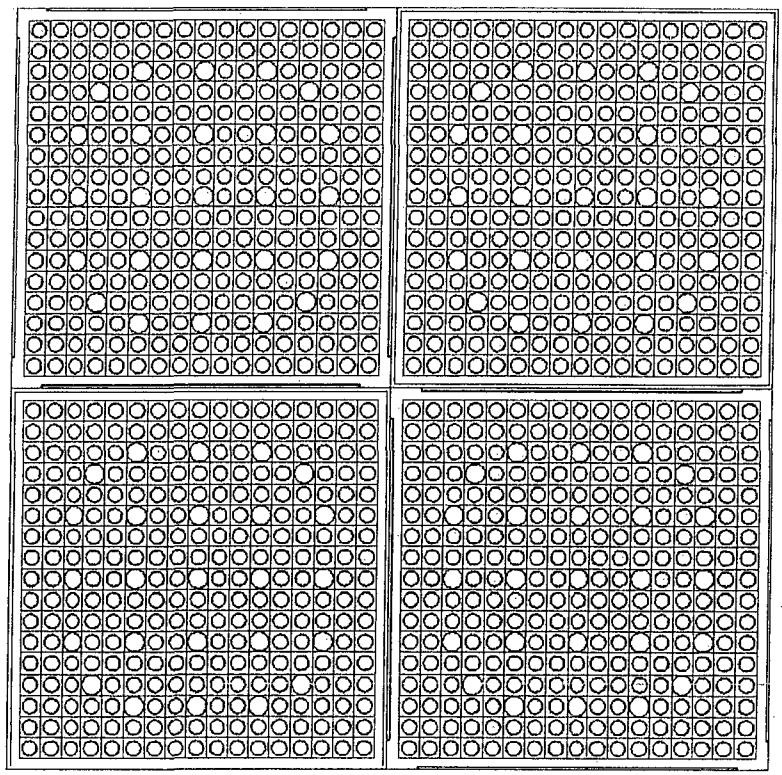

Figure 3-8. Illustration of Storage Rack Wrapper Representation

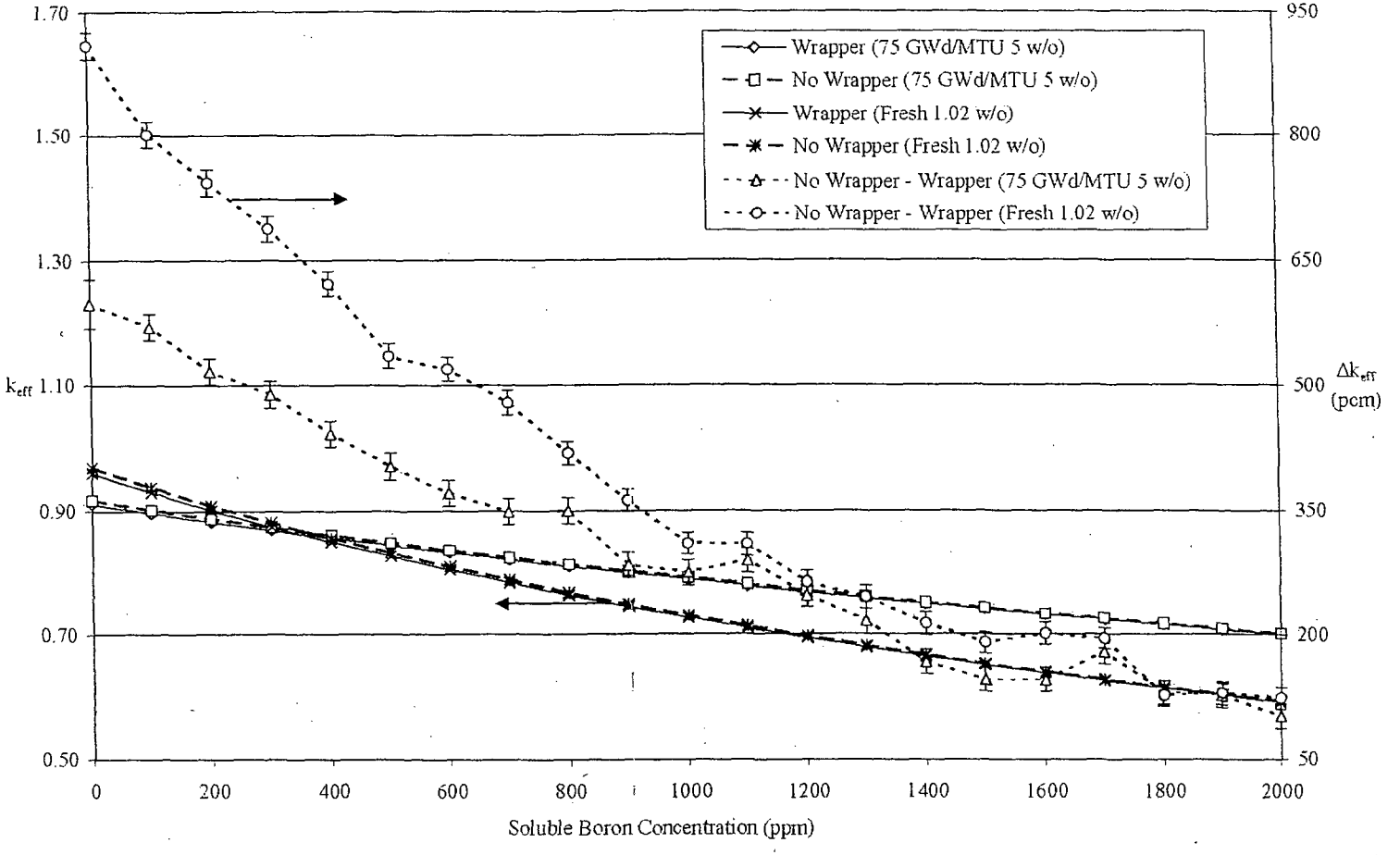

Figure 3-9. Reactivity Effects from Spent Fuel Rack Wrappers vs. Soluble Boron Concentration

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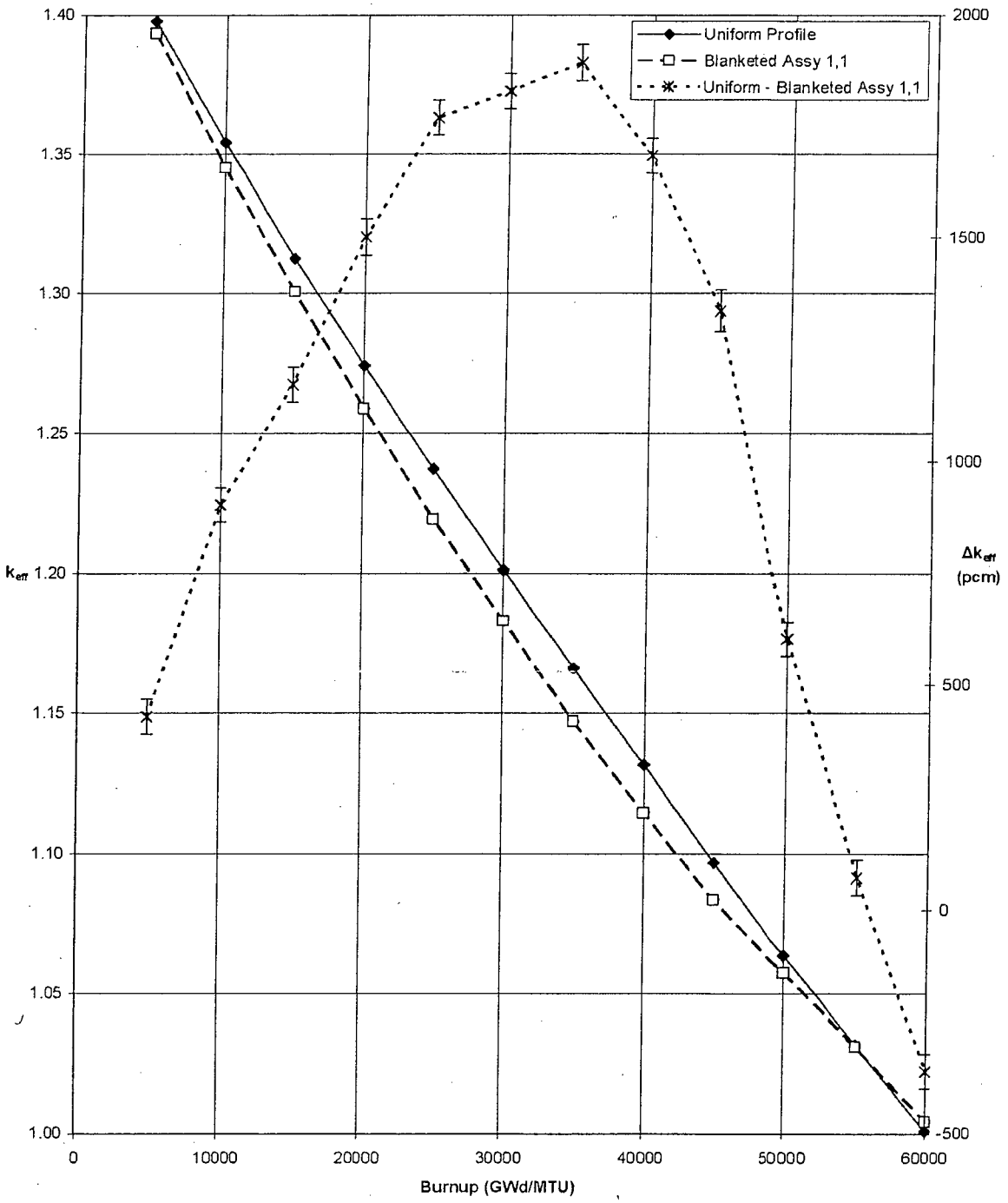

Figure 3-19. Reactivity Comparison of Axially-Blanketed, Distributed Burnup Fuel Assembly Representation vs. Unblanketed, Uniform Burnup Fuel Assembly Representation

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4.0 Clarification of Discussions in Original Analysis

The purpose of this section is to clarify various discussions presented in the original analysis to further support the criticality safety of nuclear fuel storage in the Comanche Peak Units 1 and 2 spent fuel pools.

4.1 Material Composition Modeling

The original analysis (Reference 1) states that "Standard material compositions are employed in the SCALE analyses consistent with the design input given in Section 3.0... For fresh fuel conditions, the fuel nuclide number densities are derived within the CSAS25 module using input consistent with the data of Table 2-1." The term "consistent with" may not provide the necessary level of specificity to determine the intended meaning, so a clarifying discussion is provided here.

These aforementioned statements in the original analysis intend to convey that material compositions modeled in SCALE neutronic simulations are *identical* to those shown in Table 2-1. These material specifications are chosen to represent the physical spent fuel pool storage racks and surrounding materials described in Section 3.0 of Reference 1.

Note that one exception to the composition descriptions shown in Table 2-1 involves off-nominal temperature calculations performed to determine the temperature bias for each storage configuration. In off-nominal temperature calculations, the temperature and density of H_2O is modified to appropriately reflect conditions at the temperature of interest and atmospheric (14.7 psi) conditions. The temperatures of all other materials are modified to appropriately reflect the condition of interest also.

4.2 Fuel Pellet Density

The original analysis (Reference 1) states that "Fresh and depleted fuel assemblies are conservatively modeled with a fuel stack density equal to 10.686 g/cm^3 (97.5% of theoretical UO_2 density)". To support the assumption that this treatment conservatively represents the reactivity of physical fuel assemblies, a clarifying discussion is provided here.

The as-manufactured fuel pellet density is [ ]^{a,c} of the theoretical UO_2 density of 10.96 g/cm^3. If the upper bound of this density tolerance is considered, the original analysis maintains at least [ ]^{a,c} additional fissile material relative to the physical case. Furthermore, fuel pellet dishing and chamfering are not modeled in the original analysis. As dishing and chamfering removes an additional [ ]^{a,c} of the fuel pellet volume, the original analysis maintains at least [ ]^{a,c} additional fissile material relative to the physical case.

Since the original analysis considers a fuel pellet density that is larger than the as-manufactured fuel pellets, reactivity determinations will be conservatively high relative to the physical case. This conservatism is due to the increased number of ^{235}U (and other fissile nuclides) available for fission in the neutronic models.

4.3 Enrichment Uncertainty Determination

The original analysis (Reference 1) states that "the enrichment is varied by [ ]^{a,c} at the maximum allowable fresh fuel enrichment of each storage configuration and incremental enrichments up to 5.0 w/o ^{235}U". To ensure that the computational process is clearly conveyed, it is described here in further detail.

Since the magnitude of the enrichment tolerance's effect on reactivity is a strong function of the fuel enrichment at which it is evaluated, the enrichment tolerance is assessed as a function of enrichment. [ ]^{a,c} to determine the tolerance's effect on k_{eff} of the system. The magnitude of the enrichment uncertainty is then determined by calculating the difference between the multiplication factors – each multiplication factor is pessimized by the magnitude of the Monte Carlo standard deviation associated with the simulation result. The following equation further illustrates this process.

$$\Delta k_{eff} = (k_{unc} + \sigma_{unc,\,mc}) - (k_{nom} - \sigma_{nom,\,mc})$$

Where,

Δk_{eff} is the magnitude of the enrichment uncertainty,

k_{unc} is the multiplication factor of the perturbed system,

$\sigma_{unc,\,mc}$ is the Monte Carlo standard deviation from the perturbed calculation,

k_{nom} is the multiplication factor of the nominal system,

$\sigma_{nom,\,mc}$ is the Monte Carlo standard deviation from the nominal calculation.

This calculation is then repeated for increasing values of ^{235}U enrichment that are considered in the analysis.

Note that all physical tolerance Δk_{eff} values are determined in this manner, as well as the temperature bias values.

4.4 Listing of Results in "Biases and Uncertainties Results" Tables

The original analysis (Reference 1) states that "The magnitudes of statistically significant Δk_{eff} values from manufacturing tolerances are listed in Table 4-1 through Table 4-7", and "If the reactivity contribution from a tolerance is statistically insignificant, it is neglected in the determination of biases and uncertainties." As certain physical tolerance calculations yield statistically insignificant perturbations on the k_{eff} of the system, and therefore are not listed in Tables 4-1 through 4-1, a clarifying discussion is provided here.

The original analysis determines the effects of physical tolerances on the k_{eff} of the system and summarizes the results in Tables 4-1 through 4-7. However, certain physical tolerances do not produce positive perturbations large enough to be discerned from the nominal case. In these

cases, the nominal increase in k_{eff} may be negative or smaller than the statistical combination of the each calculation's standard deviations.

If this occurs after considering a physical tolerance taken to its upper bound, an additional calculation is performed with the same tolerance taken to its lower bound. If the difference in k_{eff} remains negative or smaller than the statistical combination of the standard deviations, the tolerance's effect on reactivity may be deemed statistically insignificant. Further investigation may be performed by simulating an increased number of neutron histories in the Monte Carlo calculations to reduce the statistical variation. However, this is not required since the statistical treatment of calculation variance in the determination of total biases and uncertainties will capture the largest variance of the Monte Carlo calculations.

While the reporting of k_{eff} increases due to physical tolerances varies between storage configurations, this is only due to the differences in reactivity response to the physical tolerances outlined in Section 4.2 of Reference 1. Each storage configuration does consider all of the physical tolerances outlined in Section 4.2 of Reference 1.

4.5 Interface Requirements for the "1-out-of-4" Storage Configuration

The original analysis (Reference 1) does not consider interface requirements for the "1-out-of-4" storage configuration that is currently included in the Comanche Peak Units 1 and 2 technical specifications. While the "1-out-of-4" storage configuration does not utilize burnup credit, and need not be re-analyzed due to the stretch power uprate, it must be considered for adverse interface effects with the storage configurations analyzed in Reference 1. The interface requirement is such that an empty row of storage cells must separate the fuel assembly in a "1-out-of-4" storage configuration from any other interfacing configuration.

4.6 Axial Burnup Shape Nodalization

The original analysis (Reference 1) states that "it is required that the size of the top and bottom axial zones be small (typically 6 to 8 inches) so as to capture the steep burnup gradient with axial position", as it pertains to the axial nodalization of the fuel assembly for depletion calculations. To support the assumption that this treatment conservatively represents the reactivity of physical fuel assemblies, a clarifying discussion is provided here.

The Reference 7 analysis utilizes an eighteen-zone axial nodalization of *uniform*, 8 inch height to represent depleted fuel assemblies in criticality safety calculations. This nodalization is demonstrated in Reference 7 to be adequate to capture the reactivity "end-effect" of depleted pressurized water reactor (PWR) fuel assemblies. The original Comanche Peak Units 1 and 2 analysis from Reference 1 utilizes a four-zone axial nodalization for depleted fuel in all storage configurations except those crediting the presence of RCCAs. While four axial zones of *uniform* height would inadequately represent depleted fuel assemblies, this nodalization scheme contains three 8 inch nodes at the top of the assembly to capture the reactivity "end-effect", and a large 126 inch node to represent the remainder of the fuel assembly. This nodalization scheme is a simplified representation of that presented in Reference 7, and the zones at the top of the fuel assembly are sufficiently small to simulate the burnup gradient that produces the reactivity "end-effect". Since the middle portion of the fuel assembly depletes uniformly (as illustrated in Figure 3-10 through Figure 3-18), it is acceptable to utilize a single zone representation for this region.

Furthermore, the bottom burnup gradient may be neglected in many calculations due to its lower reactivity relative to the top of the assembly. The bottom of the fuel assembly is less reactive than the top due to the increased neutron spectral hardening at the top of the assembly due to increased moderator temperatures (and associated decreases in moderator density).

Since RCCAs may not cover the entirety of the active fuel at the bottom of the fuel assembly, storage configurations crediting the presence of RCCAs utilize a seven-zone axial nodalization for depleted fuel. Again, while seven axial zones of *uniform* height would inadequately represent depleted fuel assemblies, this nodalization scheme contains three 6 inch nodes at the top and bottom of the assembly to capture the reactivity "end-effect", and a large 108 inch node to represent the center region of the fuel assembly. Since the zones at the top and bottom of the fuel assembly are sufficiently small to simulate the burnup gradient that produces the reactivity "end-effect", any reactivity increases at the bottom of the assembly due to RCCA uncovering during spent fuel pool storage will be adequately determined.

Therefore, while the nodalization schema utilized in the original Comanche Peak Units 1 and 2 analysis differs from the recommendations in Reference 7, the use of appropriately-defined variable zone sizes ensure that the fuel assembly "end-effect" is properly represented for criticality safety calculations.

4.7 Core Operating Conditions Utilized in Depletion Calculations

The original analysis (Reference 1) states that "core operating conditions considered in all depletion calculations are representative of uprated Comanche Peak Units 1 and 2 reactor cores...the use of uprated core conditions leads to conservative determinations of reactivity." To support the assumption that this treatment is appropriate and conservatively represents the reactivity of physical fuel assemblies, a clarifying discussion is provided here.

All depletion calculations in Reference 1 utilize an uprated reactor core thermal power of 3612 MWt, and axially-varying core moderator temperatures ranging from T_{inlet} = 558.0 °F to T_{outlet} = 623.8 °F. The fuel temperatures and moderator densities corresponding to these conditions are determined internally to the PHOENIX-P depletion calculations in Reference 1. Due to the well-behaved enthalpy rise in PWR reactor cores, the axially-linear moderator temperature approximation, applied in all zoned depletion calculations, is adequate to represent the core temperature profile. The axially-uniform fuel depletion calculations utilize the core average moderator temperature of 592.8 °F. The power and temperature values are consistent with the Comanche Peak stretch power uprate licensing report.

The use of uprated core operating conditions leads to the increased production of Pu nuclides from the slightly hardened neutron spectrum resulting from increased moderator and fuel temperatures. Therefore, all fuel representations are more reactive at any given point in their depletions relative to fuel depleted at pre-uprated core operating conditions.

4.8 Temperature Bias Calculations

The original analysis (Reference 1) states that "Applicable biases factored into this evaluation are...any reactivity bias, relative to the reference analysis conditions, associated with operation

of the spent fuel pool over a temperature range of 50°F to 150°F." To ensure that the computational process is clearly conveyed, it is described here in further detail.

As reactivity effects due to moderation changes vary with the enrichment and burnup of fuel assemblies, various enrichment/burnup combinations should be investigated for their effects on the temperature bias of a storage configuration. The combinations of enrichment and burnup that are considered in the Reference 1 analysis are listed in Table 4-1. As the neutron spectrum hardens with increased fuel depletion, and therefore the fuel responds more readily to moderation changes, these combinations are chosen to utilize fuel depleted beyond the required burnup for each configuration.

Each temperature bias calculation computes the difference in keff between the temperature being investigated and the nominal (68 °F) case. The standard deviations are treated in an identical manner to that described in Section 4.3.

4.9 5% Burnup Measurement Uncertainty Justification

The original analysis (Reference 1) states that "The uncertainty in absolute fuel burnup value is conservatively calculated as 5% of the maximum fuel burnup credited in a storage configuration." To ensure that this burnup measurement uncertainty is conservative, a clarifying discussion is provided here for justification. The justification of the assembly burnup uncertainty presented here is intended to demonstrate that a value 5% is appropriate and conservative.

To demonstrate that a 5% assembly burnup uncertainty is conservative, an assembly *power* uncertainty is developed here. Such a power uncertainty conservatively will bound a burnup uncertainty due to the integration of the random variation about the mean value over the life of the assembly.

Three general terms are used to construct an overall assembly power uncertainty: the assembly power peaking uncertainty, the core power uncertainty, and the assembly loading uncertainty. The assembly power peaking and core power uncertainties may be statistically convoluted since the error in the power, both assembly peaking and absolute core power level, is likely to vary over a single cycle and from cycle to cycle. This is a valid assumption when considering an assembly burnup uncertainty because almost every assembly resides in the core for multiple cycles. The assembly loading uncertainty is considered as a bias because its value is fixed and does not vary after fabrication.

The assembly power peaking uncertainty may be derived from the $F_{\Delta H}$ uncertainty since $F_{\Delta H}$ represents the axially integrated power for a single rod. The licensed technical specification value for the $F_{\Delta H}$ uncertainty is 4% from Reference 4. This uncertainty includes a component for the uncertainty of the rod power for a given assembly power, referred to as the radial local peaking uncertainty, or pin-to-box uncertainty, of 1.24% (Reference 4). This uncertainty can be removed because the uncertainty of interest is the assembly power, so radial variations within the assembly are not of interest. Because the radial local uncertainty was statistically convoluted and increased by a 95/95 multiplier in Reference 4, the assembly power peaking uncertainty is calculated, as shown below, to be 3.32%.

$$\sigma_A = \sqrt{\sigma_{Fdh}^2 - (\sigma_{RLP} \times M_{95/95})^2} = \sqrt{4.0^2 - (1.24 \times 1.80)^2} = 3.32$$

Where,

σ_A is the assembly power peaking uncertainty,

σ_{Fdh} is the $F_{\Delta H}$ uncertainty,

σ_{RLP} is the uncertainty on radial local peaking,

$M_{95/95}$ is the appropriate 95/95 multiplier.

The core power uncertainty may be taken to be 2%. This is a typical value for the calorimetric uncertainty utilized in operations, although many plants have a significantly lower uncertainty (such as after implementing a measurement uncertainty recapture power uprate). These reduced uncertainty values are not credited here because assemblies discharged before such upgrades would not benefit from operation with a reduced uncertainty. Reference 5 provides a survey of 39 nuclear power units and provides the calorimetric uncertainty for each. A review of this data, considering only the largest reported uncertainty for instances in which multiple measurements are presented, shows the average calorimetric uncertainty to be approximately 1.6%. This justifies the use of a 2% core power uncertainty.

The uranium loading uncertainty in an assembly is conservatively assumed to be 0.2%. A review of a recent DOE/NRC Form 741 report for Westinghouse STD fuel yields a uranium loading uncertainty of 0.05%, much lower than that proposed here.

The assembly power uncertainty is subsequently determined as shown below.

$$F_{APOW} = \sqrt{(\sigma_A^2 + \sigma_{cal}^2)} + \sigma_{MTU} = \sqrt{(3.32^2 + 2.0^2)} + 0.2 = 4.1$$

Where,

F_{APOW} is the estimated assembly power uncertainty,

σ_A is the assembly power peaking uncertainty,

σ_{cal} is the calorimetric uncertainty,

σ_{MTU} is the uranium loading bias.

The calculated value of 4.1% power uncertainty provides significant conservatism to the 5% burnup uncertainty utilized in the Reference 1 criticality safety analysis. Also, the application of the 5% burnup uncertainty, in combination with the 1.0% per 30,000 MWd/MTU depletion uncertainty, in soluble boron credit calculations conservatively captures the appropriate uncertainties required in the analysis.

Note that this treatment of burnup and depletion uncertainties is also utilized in recent criticality safety analyses approved by the NRC in Reference 6.

4.10 Spent Fuel Pool Dimension Usage

The original analysis (Reference 1) describes the entire spent fuel pool models utilized to analyze interface and soluble boron credit requirements. However, this description does not mention the tolerances on spent fuel pool dimensions, as these tolerances are not considered in the analysis. To ensure that this treatment is appropriate, a clarifying discussion is provided here.

The parameters that are pertinent to the entire spent fuel pool models, and are not considered in the infinite cell models, include overall pool dimensions, minimum intra-module gap distances, and distances between the racks and pool wall. These parameters are given in Section 3.0 of the original analysis in Reference 1.

While these parameters are required to construct an entire spent fuel pool model, their tolerances will produce inconsequential impacts on desired reactivity determinations. This is, in part, due to the nature in which the models are constructed – the models utilize the minimum distances between rack modules and overall pool spacing. This approach ensures that fuel assemblies in the pool models are in as close proximity as possible, and any effects from physical tolerances are bounded. Furthermore, Monte Carlo reactivity determinations, such as those utilized in the Reference 1 analysis, do not offer the precision necessary to delineate negligible reactivity effects induced by the physical tolerances of pool parameters.

4.11 Decay Time Effects on Burnup Uncertainty

The original analysis (Reference 1) states that "^{241}Pu decay and ^{241}Am production credit is included in the burnup credit determinations", however no discussion is provided to address the effects of fuel depletion or burnup measurement uncertainties on these reactivity determinations. To support the assumption that this treatment conservatively represents the reactivity of physical fuel assemblies with accumulated decay time, a clarifying discussion is provided here.

The effects of decay time on fuel assembly reactivity are credited in the original analysis. However, only the reactivity effect from ^{241}Pu decay and associated ^{241}Am buildup is credited. While the reactivity contribution from the decay of these nuclides is substantial, and provides the largest contribution to reactivity decreases from fuel decay time, there are several other nuclides that also contribute to such a reactivity decrease. This includes the β^- decay of ^{155}Eu to ^{155}Gd, among others, that contribute to several percent of the overall reactivity decrease. The decay of such nuclides is not credited in the original Reference 1 analysis, so that the decay time credit may be ensured to be conservative.

Since the application of decay time credit is determined with isolated exponential decay calculations, the uncertainties from depletion calculations and burnup measurement are independent of its determination. Furthermore, since the depletion and burnup measurement uncertainties, along with the determination of the decay time credit, are demonstrated to contain positive criticality safety margin, it is conservative to perform decay time credit in this manner.

Table 4-1. Temperature Bias Burnup/Enrichment Combinations Investigated in Calculations

Storage Configuration	Initial Enrichment (w/o ^{235}U)	Burnup (GWd/MTU)
"4-out-of-4"	1.01	0
"4-out-of-4 with Axial Blankets"	3.0	40
	4.0	55
	5.0	65
	5.0	80
"4-out-of-4 with 1 RCCA"	5.0	70
"4-out-of-4 with 2 RCCA"	5.0	50
"3-out-of-4"	1.45	0
"3-out-of-4 with Axial Blankets"	5.0	50
"2-out-of-4"	3.55	0
	5.0	10

5.0 Summary of Results

This section presents supplementary results for the Comanche Peak Units 1 and 2 spent fuel pool criticality safety analysis with reactivity credit for burnup, RCCAs, axial blankets and ^{241}Pu decay.

5.1 Criticality Safety Margin at No Soluble Boron Conditions

The results presented in this supplementary analysis demonstrate that there exists criticality safety margin in the original Comanche Peak Units 1 and 2 analysis (Reference 1) at no soluble boron conditions. The following reactivity effects were quantified (relative to the Reference 1 analysis) at no soluble boron conditions:

- The treatment of axially-blanketed fuel assemblies with a uniform, unblanketed burnup representation exhibits a reactivity difference of -0.00361 $\pm$ 0.00039 Δk_{eff}
- The treatment of unblanketed fuel assemblies with Profile 1 axial burnup profile representation causes a reactivity difference of -0.00279 Δk_{eff} in the "4-out-of-4" storage configuration (this is independent of axially-blanketed fuel assemblies)
- The treatment of eccentric position of fuel assemblies in the "2-out-of-4" storage configuration exhibits a reactivity difference of -.00047 Δk_{eff} (this is also independent of axially-blanketed fuel assemblies)

While not explicitly quantified, the treatment of UO_2 density in the Reference 1 analysis will provide positive criticality safety margin of greater than 0.001 Δk_{eff}. This criticality safety margin, in combination with the 0.005 Δk_{eff} included as administrative margin in Reference 1, ensures that $k_{eff} < 1.0$ at no soluble boron conditions.

5.2 Criticality Safety Margin at Soluble Boron Conditions

The results presented in this supplementary analysis demonstrate that there exists criticality safety margin in the original Comanche Peak Units 1 and 2 analysis (Reference 1) at soluble boron conditions. The following reactivity effects were quantified (relative to the Reference 1 analysis) at soluble boron conditions:

- The treatment of postulated accident scenarios with soluble boron present exhibits a reactivity difference of >0.01 Δk_{eff} in all storage configurations except the "2-out-of-4"
- The treatment of postulated accident scenarios exhibits a reactivity difference of -0.00151 Δk_{eff} in the "2-out-of-4" storage configuration
- The treatment of fuel assembly grids exhibits a reactivity difference of -0.00162 $\pm$ 0.00013 Δk_{eff}

These reactivity differences, when considered in combination with the 0.005 Δk_{eff} included as administrative margin in Reference 1, ensures that $k_{eff} < 0.95$ at soluble boron conditions.

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6.0 References

1. M. Anness, "Comanche Peak Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis WCAP-16827-P, July 2007.

2. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", August 19, 1998.

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5. A. A. Wallens, "ITDP/RTDP Initial Condition Assumptions Survey – CAP IR 02-003589," LTR-SSO-02-94, November 27, 2002.

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7. J. C. Wagner and M. D. DeHart, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," ORNL/TM-2001/273, March 2003.

8. Yankee Atomic Electric Company (contributor), "PWR-AXBUPRO-SNL: Axial Burnup Profile Database for Pressurized Water Reactors", RSICC DATA PACKAGE DLC-201, September 2000.

Enclosure 3

Westinghouse authorization letter CAW-08-2451 with accompanying affidavit, Proprietary Information Notice and Copyright Notice.

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-3846
e-mail: greshaja@westinghouse.com

Our ref: CAW-08-2451

June 26, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16827-P, Addendum 1, "Supplement to Comanche Peak Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2451 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Luminant Generation Company LLC.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2451, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

R. M. Gyan/for

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: J. Thompson, NRC

CAW-08-2451

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared R. B. Sisk, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

R. B. Sisk, Manager
AP1000 Licensing & Customer Interface

Sworn to and subscribed before me
this 26th day of June, 2008

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia S. Aston, Notary Public
Murrysville Boro, Westmoreland County
My Commission Expires July 11, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, AP1000 Licensing & Customer Interface, in New Plant Projects, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component

may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16827-P, Addendum 1, "Supplement to Comanche Peak Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary), dated June 2008, for Comanche Peak Nuclear Power Plant Units 1 and 2, being transmitted by Luminant Generation Company LLC letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Comanche Peak Nuclear Power Plant Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of spent fuel pool criticality safety analysis.

This information is part of that which will enable Westinghouse to:

(a) Provide information in support of plant power spent fuel pool criticality safety analysis.

(b) Provide customer specific calculations.

(c) Provide licensing support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation associated with spent fuel pool criticality safety analysis submittals.

(b) Westinghouse can sell support and defense of the technology to its customer in the licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar information and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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