UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 10, 2009

Mr. Rafael Flores
Senior Vice President and
Chief Nuclear Officer
Luminant Generation Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 – STAFF EVALUATION AND BASIS FOR DENIAL OF LICENSE AMENDMENT REQUEST RE: REVISION TO THE TECHNICAL SPECIFICATION 3.7.17, "SPENT FUEL ASSEMBLY STORAGE" (TAC NOS. MD8417 AND MD8418)

Dear Mr. Flores:

By letter dated August 28, 2007, as supplemented by letters dated June 30 and December 11, 2008, and January 22 and February 17, 2009, Luminant Generation Company LLC (the licensee) submitted a license amendment request to revise Technical Specification (TS) 1.0, "Use and Application" and TS 3.7.17, "Spent Fuel Assembly Storage," for the Comanche Peak Steam Electric Station, Units 1 and 2. Specifically, the licensee requested to revise the rated thermal power from 3458 megawatts thermal (MWT) to 3612 MWT and for approval of the revised spent fuel pool (SFP) criticality safety analysis and the associated changes to TS 3.7.17 for revised spent fuel storage configurations. By letter dated June 27, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081510157), the U.S. Nuclear Regulatory Commission (NRC) approved the request to revise the rated thermal power level from 3458 MWT to 3612 MWT.

The NRC staff has made requests for additional information and held several discussions with your staff regarding the analysis to support the safe loading of fresh and depleted fuel assemblies in the SFP. After careful review, the NRC has concluded that your request for amending TS 3.7.17 cannot be approved. Enclosed is the NRC staff's evaluation and basis for denial of the proposed change in the amendment request. This is based on our understanding of the technical basis provided in the licensee's supplemental letters dated June 30 and December 11, 2008, and January 22 and February 17, 2009, for compliance with Title 10 of the *Code of Federal Regulations*, paragraph (b) of Section 50.68, "Criticality accident requirements," and paragraph (c)(4) of Section 50.36, "Technical specifications."

We request that your staff review the enclosed evaluation and provide comments within 15 days of the receipt of this letter. If you have any questions, please contact Balwant Singal at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosure:
Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROVIDING BASIS FOR PARTIAL DENIAL OF LICENSE AMENDMENT REQUEST

TO REVISE TECHNICAL SPECIFICATION 3.7.17

SPENT FUEL POOL CRITICALITY SAFETY ANALYSIS AND

SPENT FUEL STORAGE CONFIGURATIONS

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By letter dated August 28, 2007 (Reference 1), as supplemented by letters dated June 30 and December 11, 2008, and January 22 and February 17, 2009 (References 3, 5, 6, and 7, respectively), Luminant Generation Company LLC (the licensee) requested changes to Technical Specification (TS) 3.7.17, "Spent Fuel Assembly Storage," for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, to modify the spent fuel pool (SFP) storage configurations.

On April 17, 2008, the U.S. Nuclear Regulatory Commission (NRC) staff and the licensee and its vendor, Westinghouse Electric Company LLC (Westinghouse), held a teleconference to discuss the lessons learned from the Beaver Valley Power Station (BVPS) SFP criticality analysis amendment review, dated March 27, 2008 (Reference 2). The CPSES request was noted as being similar to the BVPS licensing action which was recently approved by the NRC. During that teleconference, the licensee identified those items it thought applied to CPSES and how the licensee planned to address them. The NRC staff also highlighted several items that applied to CPSES that the licensee had not considered as applying to its facility. One such item was the adequacy of the proposed TS. The licensee believed that this item did not apply to CPSES. The NRC staff, however, informed the licensee that the proposed change would create a conflict between TS 3.7.17 and Design Features TS 4.3.1.1. The NRC staff encouraged the licensee to consider the issues with a more critical and broad perspective. The NRC staff informed the licensee that the items discussed during the conference call need to be addressed for the staff to complete its detailed review of the licensee's request.

On June 30, 2008, the licensee submitted a supplement to its amendment request (Reference 3). On November 13, 2008, the NRC staff held a call with the licensee to clarify

Enclosure

mutual understanding of its draft request for additional information. By letter dated November 19, 2008, the NRC staff issued a request for additional information (RAI) (Reference 4). The NRC also held a call with the licensee on January 29, 2009, to discuss the status of licensee actions on the additional information requested. The licensee responded to the request for additional information in submittals dated December 11, 2008, January 22, 2009, and February 17, 2009 (References 5, 6, and 7, respectively).

The current CPSES, Units 1 and 2, SFP is divided into two Regions. Region I is a high density rack with 10.6 and 11 inch center-to-center spacing. The Region I racks are a flux-trap design with Boral neutron absorber material between the storage cells. Region II is a high density rack with a nominal center-to-center spacing of 9.0 inches. The Region II racks are a non-flux-trap design with Boraflex neutron absorber material between the storage cells. However, due to the degradation of the Boraflex material, the Boraflex is not credited in the SFP criticality analysis.

Currently, CPSES TS 3.7.17 contains four figures (Figures 3.7.17-1 through 3.7.17-4) which describe the requirements for acceptable storage of fuel assemblies in the four storage configurations allowed in the Region II SFP racks. These figures are also referenced in the Design Features TS 4.3.1.1. The requested changes for CPSES would create two new storage configurations and credit Pu^{241} decay and Am^{241} buildup. The proposed changes would be incorporated by revising the existing figures and adding two new figures (Figures 3.7.17-5 and 3.7.17-6). Each storage configuration has a geometric arrangement which must be maintained so that the SFP criticality analysis remains valid. The storage configurations may be interspersed with each other throughout the SFP, provided geometric interface requirements are met. Each storage configuration has a burnup/enrichment requirement that must be met for a fuel assembly to be stored in that configuration.

The analysis was performed for the licensee by Westinghouse as WCAP-16827-P, Revision 0, and was submitted as Enclosure 3 to Reference 1. Addendum 1 to WCAP-16827 was submitted by the licensee as Enclosure 1 to Reference 3. The licensee provided responses to the NRC staff RAI by letters dated December 11, 2008, January 22, 2009, and February 17, 2009 (References 5, 6, and 7, respectively).

2.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria for Nuclear Power Plants," Criterion 62, "Prevention of criticality in fuel storage and handling," requires,

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

In Section 50.36, "Technical specifications," of 10 CFR, the Commission established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS.

Paragraph 50.36(c)(4), "Design features," of 10 CFR, requires,

> Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

Paragraph 50.68(b)(1) of 10 CFR requires,

> Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

Paragraph 50.68(b)(4) of 10 CFR requires,

> If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The revised CPSES SFP criticality analysis takes credit for soluble boron. Therefore, the regulatory requirement is for the CPSES SFP k-effective (k_{eff}) to remain below 1.0, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water and below 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water. The revised CPSES SFP criticality analysis uses an analytic acceptance criterion of k_{eff} <0.995, if flooded with unborated water. The revised CPSES SFP criticality analysis does not explicitly state a specific analytic acceptance criterion if flooded with borated water. As performed in the licensee's analysis the borated water condition uses an analytic acceptance criterion of k_{eff} <0.945 if flooded with borated water. Thus, the analysis reserved 0.005 Δ k_{eff} margin for each criterion.

The revised CPSES SFP criticality analysis requires specific geometric arrangements of fuel assemblies within the SFP to meet the 10 CFR 50.68(b)(4) regulation. Therefore, there is a regulatory requirement for those geometric arrangements to be captured in the TS Section 4.0, "Design Features," of the TSs. This is being accomplished by citing the figures from LCO 3.7.17 in the TS 4.3.1.1 of the CPSES, Units 1 and 2 TSs. In determining whether or not a criticality code validation is adequate, the NRC staff relies on NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," dated January 2001 (Reference 11).

3.0 TECHNICAL EVALUATION

3.1 Proposed Change

The current CPSES TSs contain a 4-of-4 storage configuration in which all of the fuel assemblies in each repeating 2x2 array must meet the burnup/enrichment loading curve in Figure 3.7.17-1. This figure does not consider Pu^{241} decay and Am^{241} buildup. The proposed change will replace the current Figure 3.7.17-1 with new Figure 3.7.17-1 that does consider Pu^{241} decay and Am^{241} buildup. In addition, the amount of burnup required for each enrichment is significantly increased. The proposed change also includes a 4-of-4 storage configuration in which all of the fuel assemblies in each repeating 2x2 array must have axial blankets and meet a new burnup/enrichment loading curve, which also considers Pu^{241} decay and Am^{241} buildup. The proposed change will also provide new Figure 3.7.17-2 for the '4-of-4 with Axial Blankets' storage configuration.

The current CPSES TS contain a 3-of-4 storage configuration in which one of the storage cells in each repeating 2x2 array must be empty while the others contain fuel assemblies which meet the burnup/enrichment loading curve in the current Figure 3.7.17-2. This figure does not consider Pu^{241} decay and Am^{241} buildup. The proposed change will replace the current Figure 3.7.17-2 with a new Figure 3.7.17-3 that does consider Pu^{241} decay and Am^{241} buildup. In addition, the amount of burnup required for each enrichment is significantly increased. The proposed change also includes a 3-of-4 storage configuration in which one of the storage cells in each repeating 2x2 array must be empty while the others contain fuel assemblies which have axial blankets and meet a new burnup/enrichment loading curve, which also considers Pu^{241} decay and Am^{241} buildup. The proposed change will also provide new Figure 3.7.17-4 for the '3-of-4 with Axial Blankets' storage configuration.

The current CPSES TS contain a 2-of-4 storage configuration (checkerboard) in which two of the storage cells in each repeating 2x2 array must be empty while the others contain fuel assemblies which meet the burnup/enrichment loading curve in the current Figure 3.7.17-3. This figure does not consider Pu^{241} decay and Am^{241} buildup. The proposed change will replace the current Figure 3.7.17-3 with a new Figure 3.7.17-5 that does consider Pu^{241} decay and Am^{241} buildup. In addition, the amount of burnup required for each enrichment is significantly decreased.

The current CPSES TS also contain a 1-of-4 storage configuration in which three of the storage cells in each repeating 2x2 array must be empty while the other may contain fresh fuel. This storage configuration is depicted in the current Figure 3.7.17-4. The current Figure 3.7.17-4 also contains the interface requirements for the storage configurations. The current Figure 3.7.17-4 would be replaced with a new Figure 3.7.17-6; the only difference is incorporation of the new figure numbers.

The figures are located in TS 3.7.17 and are referenced in TS 4.3.1.1. The original submittal included the above-described changes to the figures in TS 3.7.17, but did not revise the description of the figures in TS 4.3.1.1. As mentioned before this was brought to the licensee's attention during the April 17, 2008, teleconference. However, the June 30, 2008, supplement did not resolve the conflict. The final RAI response submitted on February 17, 2009 (Reference 7), provided an appropriate correction to TS 4.3.1.1.

3.2 Methodology

There is no generic methodology for performing SFP criticality analyses. The NRC withdrew its approval of WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996 (Reference 8), on July 27, 2001 (Reference 9). In its withdrawal letter the NRC stated, "For future licensing actions, licensees will need to submit plant-specific criticality calculations for spent fuel pool configurations that include technically supported margins."

The NRC staff issued a memorandum on August 19, 1998, containing guidance for performing the review of SFP criticality analysis (Reference 10). This memorandum is known colloquially as the 'Kopp Letter,' after the author. While the Kopp Letter does not specify a methodology, it does provide some guidance on more salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), borated and unborated. The Kopp Letter has been used as a touchstone for numerous PWR SFP criticality analyses since and including this CPSES analysis. The guidance in the Kopp Letter can be summarized as: determine the biases and uncertainties for the parameters affecting reactivity in the SFP and apply them in the conservative direction. The guidance allows for the statistical combination of uncertainties, provided they are independent.

3.3 Computer Code Validation

WCAP-16827-P, employs the following software: (1) SCALE version 4.4, with the SCALE version 4.4 versions of the 44- and 238-group Evaluated Nuclear Data File Version 5 (ENDF/B-V) neutron cross section libraries, and (2) the two-dimensional transport lattice code PHOENIX-P, with an Evaluated Nuclear Data File Version 6 (ENDF/B-VI) neutron cross section library. SCALE is utilized for reactivity determinations of fuel assemblies in the CPSES, Units 1 and 2, SFPs. The PHOENIX-P code is used for simulation of in-reactor fuel assembly depletion.

Validation of SCALE 4.4, for purposes of fuel storage rack analyses, is based on the analysis of 30 selected critical experiments from two experimental programs: 19 from the Babcock & Wilcox experiments in support of "Close Proximity Storage of Power Reactor Fuel" and 11 from the Pacific Northwest Laboratory (PNL) Program in support of the design of "Fuel Shipping and Storage Configurations." The results of benchmarking these 30 experiments are provided in WCAP-16827-P and results in a code bias and uncertainty.

The experiments used for validation are the same experiments Westinghouse has used to validate criticality codes in the past. The validation methodology submitted in WCAP-16827-P lacks key elements of a validation analysis. In particular, the area of applicability evaluation and trend analysis are not included. Consequently, more information is needed for the docketed validation to be considered complete.

In addition to using the SCALE 4.4 code to perform the criticality analyses, the licensee employed the PHOENIX-P code to perform the fuel depletions used in the analysis. PHOENIX-P is a two-dimensional, multi-group transport theory lattice code. The multigroup cross sections are based on ENDF/B-VI. PHOENIX-P performs a two-dimensional 70-group nodal flux calculation which couples the individual sub-cell regions (pellet, cladding, and moderator) as well as surrounding rods via a collision probability technique.

WCAP-16827-P, Addendum 1 (Reference 3), employs the following software: (1) SCALE5.1 and (2) PARAGON software. The substitution was made to ease the computational burden associated with the methods utilized in WCAP-16827-P. SCALE 5.1 and PARAGON are utilized as direct replacements for SCALE4.4 and PHOENIX-P, respectively. WCAP-16827-P, Addendum 1, indicates that a validation has been performed and lists a code bias and uncertainty. However, information regarding that validation was not provided. Therefore, the NRC staff could not evaluate the adequacy of the validation.

The licensee's RAI responses are largely silent on what code was used in the analysis, with only one response indicating which codes were used (i.e., the response to RAI #15 indicates SCALE 5.1 and PARAGON were used to determine the effect of the depletion parameters). As such, it appears that SCALE 5.1 and PARAGON were used for all of the RAI responses related to the reasons stated in WCAP-16827-P, Addendum 1. This results in an apparent mix and match of the different codes. The response to RAI #17 recalculates the 'enrichment uncertainty,' 'burnup measurement uncertainty,' and 'depletion uncertainty' without the presence of soluble boron in the SFP. These recalculations are presumably performed with SCALE 5.1 and PARAGON, yet the SCALE 4.4 methodology bias and uncertainty along with the other biases and uncertainties determined with SCALE 4.4 are used in the Total Biases and Uncertainties. The effect of the apparent mixing the two methods is not addressed.

In determining whether or not a criticality code validation is adequate, the NRC staff relies on NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," dated January 2001 (Reference 11). Using NUREG/CR-6698, the NRC staff concludes that the submittal lacks the necessary key elements of a validation analysis. Without these elements, there is insufficient information for the staff to determine whether or not the validations of the criticality codes used in this license amendment request are adequate.

3.4 Spent Fuel Characterization

For the SFP criticality analysis, the fuel must be characterized appropriately. Characterization of fresh fuel is relatively is based primarily on U^{235} enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, or are bounded by values used in the analysis. These tolerances and bounding values would also carry through to the spent fuel and the standard practice has been to treat the uncertainties as unaffected by the depletion. The characterization of spent fuel is more problematic. Spent fuel characterization is based on the specifics of its initial conditions and operational history in the reactor. That characterization has three main areas: a burnup uncertainty, the axial apportionment of the burnup, and the core operation that achieved that burnup.

3.4.1 Burnup Uncertainty

In the Kopp Letter, the NRC staff provided guidance for determining the depletion uncertainty, stating that:

> A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

The 5 percent reactivity decrement has been used throughout the industry since the issuance of the Kopp Letter on August 19, 1998.

Rather than use the 5 percent reactivity decrement as the depletion uncertainty, the WCAP-16827 analysis used a 5 percent decrease in the burnup of interest. WCAP-16827 does not identify the deviation from NRC staff guidance, nor does it provide any justification for using this method of calculating the uncertainty. The NRC staff has previously rejected the use of the '5 percent decrease in the burnup of interest' methodology for calculating the depletion uncertainty.

During the April 17, 2008, teleconference, the licensee's vendor stated there was new information to support the '5 percent decrease in the burnup of interest' methodology for calculating the depletion uncertainty that the NRC staff had not yet been provided. However, in the supplement, the rationale provided for the '5 percent decrease in the burnup of interest' methodology for calculating the depletion uncertainty was taken almost verbatim from an earlier submittal the NRC staff had rejected. During the teleconference, it was apparent that the 'depletion uncertainty' in the original WCAP-16827-P was in actuality a 'burnup measurement uncertainty' and an actual 'depletion uncertainty' was not included in the unborated portion of the analysis. A 'depletion uncertainty' was included in the borated portion of the analysis, but that 'depletion uncertainty' did not comply with the NRC staff guidance either. Consequently, the NRC staff suggested the licensee to recalculate the 'depletion uncertainty' using the '5 percent of the reactivity decrement' methodology and to use it in both the unborated and borated analyses.

In its RAI response, the licensee indicated that a 'depletion uncertainty' using the '5 percent of the reactivity decrement' methodology is included in the rack-up of the biases and uncertainties for all of the storage configurations the licensee has requested to be incorporated in the TS. There was no information on what assumptions and input parameters were used in this portion of the analysis. If non-conservative core operating parameters or non-conservative axial burnup profiles were used in the analysis, then the 'depletion uncertainty' would be non-conservative. For example, the axial burnup profile RAI response, for the axial burnup profile used for the "4-out-of-4 with Axial Blankets" storage configuration, indicates there is a significant non-conservatism, depending on which axial burnup profile is used, but it is unclear which profile was used in this portion of the analysis. The NRC staff, therefore, concluded that there was insufficient information for the NRC staff to evaluate the values used in the analysis.

The RAI response also included a revised lower 'burnup measurement uncertainty.' The NRC staff makes no evaluation of the 'burnup measurement uncertainty,' except to note that for the "4-out-of-4" storage configuration, the value for the 5.0 weight percent (w/o) U^{235} enrichment is less than the 4.0 w/o U^{235} enrichment value. Also, out of the eight times a 'burnup measurement uncertainty' is determined throughout the RAI responses, this is the only instance where a higher enrichment has a lower value.

3.4.2 Axial Apportionment or Burnup Profile

Another important aspect of fuel characterization is the selection of the burnup profile. At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis has shown that this results in an under prediction of k_{eff} and, generally, the under prediction becomes larger as burnup increases. This is what is known as the 'end effect.' Judicious selection of the axial burnup profile is necessary to ensure k_{eff} is not under predicted due to the end effect. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," dated March 2003 (Reference 12), provides insight for selecting an appropriate axial burnup profile.

For non-blanketed fuel assemblies, WCAP-16827 used what it characterized as the limiting axial burnup profile:

> Input to this analysis is based on a limiting axial burnup profile data provided in the DOE (Department of Energy) Topical Report, as documented in Reference 12.

Reference 12 in WCAP-16827 is Oak Ridge National Laboratories (ORNL) report ORNL/TM-1999/246, "Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations," dated March 2000 (Reference 13). While this ORNL report discusses distributed axial burnup profiles, it does not identify a single 'limiting axial burnup profile.' Neither does the source of the axial burnup profile data: DOE Topical Report DOE/RW-0472, Revision 2, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," dated September 1998 (Reference 18). In actuality, the axial burnup profile used in WCAP-16827 is not listed in ORNL/TM-1999/246. However, it does correspond to an axial burnup profile listed in NUREG/CR-6801. The axial burnup profile used in WCAP-16827-P is indicated by NUREG/CR-6801 to be non-conservative for burnups below 46 gigawatt day per metric ton unit (GWD/MTU), while WCAP-16827-P used this profile exclusively for any case involving a distributed axial burnup profile. Therefore, cases involving a distributed axial burnup profile at a burnup below 46 GWD/MTU are potentially non-conservative. NUREG/CR-6801 is generic, considering the axial burnup profiles for several fuel design types, therefore, allowing for a more specific analysis to show acceptable results. The licensee and its vendor attempted to do that in WCAP-16827-P, Addendum 1. The NRC staff had some concerns with that analysis. In particular, the analysis focused only on one enrichment in one storage configuration.

The simulations of axial blankets of 2.4 w/o U^{235} in the "4-out-of-4" storage configuration in WCAP-16827-P, Addendum 1, with a different distributed axial burnup profile from the database, appeared to show a "...maximum reactivity increase...." However, Figure 3-5 in WCAP-16827-P, Addendum 1, shows increasing non-conservatism with decreasing burnup. Additionally, the non-conservative simulations only go down to a burnup of 30 GWD/MTU while the Figure 3.1, "2.0 w/o ^{235}U" burnup credited in the "4-out-of-4" storage configuration goes down to 18 GWD/MTU and the simulations used to determine that credit go down to 15 GWD/MTU. This appears to indicate that the WCAP-16827-P, Addendum 1, stated "...maximum reactivity increase..." was probably not the maximum reactivity increase. The "...maximum reactivity increase..." was to be used as an uncertainty for one enrichment in one storage configuration despite the indications that the non-conservatism was systemic.

To address this issue in the RAI response, the licensee appears to have limited the analysis. In the RAI response, the licensee indicates that the only non-blanketed fuel assemblies are those of relatively low enrichment and burnup. Another case study was performed. The analysis was limited to axial blankets of 3.4 w/o ^{235}U fuel assemblies with site-specific axially distributed burnup profiles. The new case study indicates there is considerable conservatism in the original analysis. However, the RAI response contains the following informational deficiencies:

- There is insufficient information that would enable the NRC staff to evaluate the site-specific axially distributed burnup profiles used in the RAI response.
- The storage configuration in which the case study was performed is unknown to the staff since it is not stated explicitly in the RAI response.
- Since this case study is limited to one enrichment, there is no information regarding applicability to lower enrichments.
- The Addendum appears to show a significant non-conservatism, and the RAI response showed a significant conservatism. Since information is lacking regarding the input parameters for either case, the NRC staff cannot determine whether or not the conservatism indicated in the RAI response is valid or not.
- The TS figures governing the non-blanketed fuel assemblies were not revised to reflect the maximum enrichment the analysis now covers.

WCAP-16827-P uses a uniform burnup profile to model fuel assemblies with axially blanketed fuel. WCAP-16827-P cited NUREG/CR-6801 as justification. With respect to representing the burnup profile of an assembly with axial blankets, the NRC staff is unaware of any generic analysis that would support definitive conclusions. While NUREG/CR-6801 does state in its conclusion that, "...the axial blankets have significantly lower enrichment than the central region, the end effect for assemblies with axial blankets is typically very small or negative. Furthermore, the lower the initial enrichment of the axial blankets is with respect to the higher enrichment central region, the lower is the end effect." To put 'very small' in context, NUREG/CR-6801 considers 0.005 Δk_{eff} to be small. Since WCAP-16827-P reserves an analytical margin of 0.005 Δk_{eff}, a 'very small' impact is worth determining. WCAP-16827-P, Addendum 1, provides some information about this item. WCAP-16827-P, Addendum 1, provides a comparison between a uniform profile and a limiting axial profile selected from profiles of actual blanketed fuel assemblies from recent CPSES cores.

The NRC staff had several concerns with this analysis. In particular, the staff was concerned with the following statement from WCAP-16827-P, Addendum 1:

> ...the representation is most conservative at 35 GWD/MTU, reaching a maximum reactivity difference of 1893 ± 41 pcm [percent-millirho] Δkeff(1 pcm = 10^{-5}). The least conservative time of life is at 60 GWD/MTU when the reactivity difference is 361 ± 39 pcm Δkeff.

The comparison is provided in Figure 3-19 which shows the 'least conservative time of life' to be a negative difference, which is a non-conservative rather that a "least conservative" resultant. Figure 3-19 also indicates the non-conservatism becomes larger with increasing burnup. Since the "4-out-of-4 with Axial Blankets" storage configuration credits 62,662 megawatt days per metric ton unit (MWD/MTU) of burnup for 5.0 w/o enriched U^{235} with zero Pu^{241} decay time, this point is non-conservative by an amount that is reasonably expected to exceed the WCAP-16827-P reserved analytical margin of 0.005 Δk_{eff}. Therefore, the "4-out-of-4 with Axial Blankets" storage configuration did not appear to meet the requirements of 10 CFR 50.68 based on this consideration alone. Additionally, as that point is used to determine a second order polynomial for controlling the burnup/enrichment loading curve for the "4-out-of-4 with Axial Blankets" storage configuration for zero Pu^{241} decay time, the equation is non-conservative by some amount. As other simulations in the "4-out-of-4 with Axial Blankets" storage configuration determinations use burnup levels at or above 60 GWD/MTU, those are also non-conservative by some amount.

To address this issue in the RAI response, the licensee limited the analysis. The RAI response includes case studies based on blanket enrichments of natural U^{235}, 2.0 w/o U^{235}, and 2.6 w/o U^{235}. The case studies also use different site-specific distributed axial burnup profiles from each blanket enrichment category. The case studies are performed for initial enrichments of 4.0 w/o U^{235} and 5.0 w/o U^{235}. The new case studies indicate there is considerable conservatism in the natural U^{235} and 2.0 w/o U^{235}, blankets. The new case studies indicate there is considerable non-conservatism for the 2.6 w/o U^{235} blankets. The non-conservatism is to be incorporated as a bias. However, the RAI response contains the following informational deficiencies:

- There is insufficient information that would enable the NRC staff to evaluate the site-specific axially distributed burnup profiles used in the RAI response.

- The storage configuration in which the case study was performed in the "4-out-of-4 with Axial Blankets" storage configuration. While the RAI response evaluates the "3-out-of-4 with Axial Blankets" storage configuration for 2.6 w/o U^{235} blankets, no apparent case is made for the other blankets in the "3-out-of-4 with Axial Blankets" storage configuration.

- The Addendum showed a significant non-conservatism, and the RAI response showed a significant conservatism. Since there is no information regarding the input parameters for either case, the NRC staff cannot determine whether or not the conservatism indicated in the RAI response is valid or not.

3.4.3 Burnup History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR [Light-Water Reactor] Fuel," dated February 2000 (Reference 14), provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on criticality analysis in storage and transportation casks, the basic principles with respect to the depletion analysis apply generically to SFPs, since the phenomena occur in the reactor as the fuel is being used. The results have some translation to SFP criticality analyses, especially when the discussion includes the effect in an infinite lattice analysis, similar to that performed for SFP analyses. The basic premise is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum Pu^{241} production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in maximum Pu^{241} production. Neither the initial submittal in Reference 1 nor the supplement in Reference 3 addressed all of the parameters. The RAI responses addressed all of the parameters. However, the RAI responses determined that it was conservative to ignore the presence of Westinghouse Integral Fuel Burnable Absorbers (IFBAs) and Wet Annular Burnable Absorbers (WABAs). This conclusion is contradictory to that reached in NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," dated March 2002 (Reference 15), and NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," dated March 2002 (Reference 16), and other SFP licensing activities. Therefore, the NRC staff could not substantiate the basis for the licensee's conclusion.

3.5 Determination of Soluble Boron Requirements

The regulatory requirement for soluble boron credit is that the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water.

Neither the initial submittal in Reference 1 nor the supplement in Reference 3 adequately addressed the determination of the soluble boron requirements. The NRC staff's concerns are highlighted in RAI #26 (Reference 4), and, therefore, are not replicated in their entirety in this safety evaluation.

The RAI response recalculates the 'enrichment uncertainty,' 'burnup measurement uncertainty,' and 'depletion uncertainty' assuming the presence of soluble boron in the SFP. The 'enrichment uncertainty' increased while the 'burnup measurement uncertainty' and 'depletion uncertainty' decreased, with the net result an indicated overall decrease in the total rack-up of biases and uncertainties. However, the NRC staff has the following concerns with the information provided:

- Other biases and uncertainties are not recalculated assuming the presence of soluble boron in the SFP. The NRC staff remains concerned about the effect that the large amount of soluble boron credited in the analysis will have on the temperature bias.

- There is insufficient information on what assumptions and input parameters were used in this portion of the analysis. If non-conservative core operating parameters were used in the depletion, then the 'depletion uncertainty' would be non-conservative. If non-conservative axial burnup profiles were used in the analysis, then the 'depletion uncertainty' would be non-conservative.

- The RAI response states,

> Blanketed configurations were not considered here even though they produced slightly lower unborated Target k_{eff} values in the response to question 17. This is more than offset by the increased burnup limits associated the unblanketed fuel.

  The NRC staff concludes that this is inconsistent with the information provided in the response to RAI #17. Due to the significant penalty imposed on the "4-out-of-4 with Axial Blankets" storage configuration determined in the response to RAI #12, the "4-out-of-4 with Axial Blankets" storage configuration unborated target k_{eff} values are significantly lower than those of the "4-out-of-4" storage configuration.

As identified in the supplement (Reference 3) and the response to RAI #18 (Reference 5), there is a potential non-conservatism for neglecting fuel assembly grids when crediting soluble boron. This potential non-conservatism is not addressed in the licensee's soluble boron crediting analysis.

3.6 <u>Determination of Storage Configuration Interface Requirements</u>

WCAP-16827, Section 4.5.1, describes the analysis that was performed to determine the acceptable interface between the various storage configurations. Since the analysis calculates a separate Target k_{eff} for each burnup/enrichment combination in the storage configuration, each storage configuration will have sub-configurations based on the individual Target k_{eff}. The interface model uses one sub-configuration from each of the two storage configurations being compared. The model used was a spent fuel model where one storage module was filled with one storage configuration sub-configuration and the other eight modules were filled with a sub-configuration from a different storage configuration. The criterion compares the k_{eff} of the interface model to the largest Target k_{eff} of the two storage configuration sub-configurations used in the model. If the interface model k_{eff} is the lesser value in that comparison, then the interface is deemed acceptable. Neither the original WCAP-16827 nor Addendum 1 explicitly considered all of the storage configurations. Those configurations that were originally left out were included in the response to RAI #6 in Reference 5. Based on the information provided, the NRC staff is unable to conclude that the analysis demonstrates reasonable assurance that the storage configuration interface requirements would support a regulatory determination that requirements are met.

- The acceptance criteria used in the submittal compares the k_{eff} of the interface model, which includes radial leakage, with a storage configuration Target k_{eff}, which does not include radial leakage. This appears to prejudice the comparison in favor of the interface model, as it would be expected to have a lower k_{eff} than a model without axial leakage.

- The acceptance criteria of using the largest Target k_{eff} of the two storage configuration sub-configurations used in the model also appears to prejudice the comparison in favor of the interface model. NUREG/CR-6683, "A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage," dated September 2000 (Reference 17), indicates that the reactivity of the most reactive storage configuration will be reduced by the less reactive storage configuration, while the reactivity of the less reactive storage configuration will be increased. Therefore, the comparison should be with the less reactive storage sub-configuration.

- The two sub-configurations used in the interface model are not justified sufficiently, as to whether or not they create bounding conditions. In keeping with NUREG/CR-6683, the bounding cases would likely involve the maximum Target k_{eff} sub-configuration from one storage configuration and the minimum Target k_{eff} sub-configuration from a different storage configuration.

- There is insufficient information that indicates whether or not other modeling changes exist that would prejudice the comparison in favor of the interface model (i.e., it is not stated which storage configuration is in the one module and which is in eight), and it is not stated whether or not intra-module spacing is included.

- The interface issue does not appear to have been reexamined with the revised burnup/enrichment combinations and Target k_{eff} values from the responses to other RAIs.

The introduction of sub-configurations within a storage configuration also introduces the interface question within a storage configuration. In keeping with NUREG/CR-6683, if the maximum Target k_{eff} sub-configuration from one storage configuration and the minimum Target k_{eff} sub-configuration from the same storage configuration are placed next to each other, then the reactivity of the sub-configuration with the minimum Target k_{eff} would be expected to increase. That increase may be sufficient such that the sub-configuration with the minimum Target k_{eff} would not meet the regulatory requirements.

4.0 CONCLUSION

Since the licensee takes credit for soluble boron, the regulatory requirement is taken from 10 CFR 50.68(b)(4) which requires,

> If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated

water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The licensee used an analytical acceptance limit of $k_{eff} < 0.995$ for the unborated condition and an analytical acceptance limit of $k_{eff} < 0.945$ for the borated condition. Thus, the analysis reserved a 0.005 Δ k_{eff} margin. After reviewing the licensee's August 28, 2007, submittal (Reference 1), and supplements, the NRC staff could not conclude definitively whether the proposed TS change would meet the regulatory requirements of 10 CFR 50.68(b)(4) and 10 CFR 50.36(c)(4). In some cases, the supporting analysis indicated the proposed TS would not meet the regulatory requirements and, therefore, the NRC staff requested additional information (Reference 4). The additional information provided by the licensee did not resolve the NRC staff's concerns and, therefore, for the reasons identified above, the NRC staff has proposed to deny the amendment request.

5.0 REFERENCES

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2. N. Morgan, U.S. Nuclear Regulatory Commission letter, to P. Sena, III, FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, "Beaver Valley Power Station Unit 2 - Issuance of Amendment RE: Incorporation of the Results of a New Spent Fuel Pool Criticality Analysis (TAC No. MD2378)," dated March 27, 2008 (ADAMS Accession No. ML080730476).

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4. B. Singal, U.S. Nuclear Regulatory Commission, to Mr. M. Blevins, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 – Request for Additional Information Regarding Spent Fuel Pool Criticality License Amendment Request (TAC Nos. MD8417 and MD8418)," dated November 19, 2008 (ADAMS Accession No. ML083120002).

5. M. Blevins, Luminant Power, to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Regarding Spent Fuel Pool Criticality License Amendment Request (TAC Nos. MD8417 and MD8418)," TXX-08148, dated December 11, 2008 (ADAMS Accession No. ML083570151).

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8. WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.

9. S. Dembeck, U.S. Nuclear Regulatory Commission, to H. A. Sepp, Westinghouse Electric Company, "Non-Conservatisms in Axial Burnup Biases for Spent Fuel Rack Criticality Analysis Methodology," dated July 27, 2001 (ADAMS Accession No. ML012080337).

10. U.S. Nuclear Regulatory Commission, Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998 (ADAMS Accession No. ML003728001).

11. Science Applications International Corporation/U.S. Nuclear Regulatory Commission, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, dated January 2001 (ADAMS Accession No. ML050250061).

12. Oak Ridge National Laboratory/U.S. Nuclear Regulatory Commission, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," NUREG/CR-6801, dated March 2003 (ADAMS Accession No. ML031110292).

13. J. C. Wagner and M. D. DeHart, Oak Ridge National Laboratory, "Review of Axial Burnup Distribution Considerations for Burnup Credit Calculations," ORNL/TM-1999/246, dated March 2000.

14. Oak Ridge National Laboratory/U.S. Nuclear Regulatory Commission, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, dated February 2000 (ADAMS Accession No. ML003688150).

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18. U.S. Department of Energy, "Topical Report on Actinide-Only Burn-up Credit for PWR Spent Nuclear Fuel Packages," DOE/RW-0472, Revision 2, dated September 1998 (ADAMS Accession No. ML070780665).

Principal Contributor: Kent Wood

Date: July 10, 2009

We request that your staff review the enclosed evaluation and provide comments within 15 days of the receipt of this letter. If you have any questions, please contact Balwant Singal at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosure:
Safety Evaluation

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ADAMS Accession No. ML091870479 ***SE memo dated** ****via email**

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