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1. For the limiting Group F2 fuel assembly selection, the only design that does not include axial blankets was not included. The licensee stated that "the combination of this fuel type's [[ ]] it to be bounded by [[ ]]." Please list the parameters that caused this assembly to not be considered.

Response:

[ ]^{a,c} The differences between the unblanketed fuel assembly and the Group F2 design basis assembly are outlined in Table 1-1.

Table 1-1: Comparison of the Unblanketed and Design Basis Fuel Assembly Designs

Parameter	Unblanketed Design	Group F2 Design Basis
Blanket Enrichment[1] (wt% ^{235}U)	1.6, 2.4, and 3.1	[ ]^{a,c}
95/95 Fuel Density (%TD)	[ ]^{a,c}	[ ]^{a,c}
Burnable Absorber Loading	20 WABA fingers	[ ]^{a,c}
Reactor Power (MW_{th})	3458	3612

Notes:
1. The blanket enrichment refers to the enrichment of the top and bottom 6" of both the blanketed and unblanketed assembly.

Unblanketed fuel assemblies were only made in enrichments of 1.6, 2.4, and 3.1 weight percent (wt%) ^{235}U. [ ]^{a,c} The parameters used for the unblanketed fuel depletion calculations are outlined in Table 1-2.

Table 1-2: Depletion Parameters for Unblanketed Assembly Design

Parameter	Unblanketed Design	Unblanketed Depletion Parameter
Reactor Power (MW_{th})	3458	[ ]^{a,c}
95/95 Fuel Density (%TD)	[ ]^{a,c}	[ ]^{a,c}
Assembly Temperature Profile[1]	[ ]^{a,c}	[ ]^{a,c}
Burnable Absorber Loading	20 WABA fingers	[ ]^{a,c}

Notes:
1. [ ]^{a,c}

The spent fuel isotopics generated in the depletion calculations were imported to KENO V.a and the reactivity of the unblanketed fuel assemblies were calculated. The reactivity of the unblanketed fuel assembly was compared to the reactivity of the F2 design basis assembly. The results of the reactivity comparison for burnup bin 1 are provided in Table 1-3.

Table 1-3: Reactivity Comparison at 3.0 wt% ^{235}U for Burnup Bin 1 a,c

[ ]

The results in Table 1-3 show that the Group F2 design basis assembly is more reactive than the unblanketed fuel assembly in burnup bin 1. [

]a,c

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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2. In the selection of the limiting Group F2 assembly, the licensee explained that the combination of Wet Annular Burnable Absorber (WABA) and Integral Fuel Burnable Absorber (IFBA) conservatively bounds the other designs that only use one or the other (i.e., WABA or IFBA). Please provide the results of the analysis that demonstrates this is true.

Response:

Westinghouse has provided below a reactivity comparison of the design basis burnable absorber loading to the burnable absorber loadings covered in Table 6-16 of Reference 1. [

]a,c

Past Use of the 64 IFBA/24 WABA Burnable Absorber Loading

[

]^{a,c}

Table 2-1: Reactivity Comparison of the Design Basis and 64 IFBA/24 WABA Past Use Assemblies

a,c

[

]^{a,c}

[

]a,c

a,c

Table 2-2: Reactivity Comparison of the Design Basis and 64 IFBA/24 WABA Future Use Assemblies

[

]^{a,c}

a,c

Table 2-3: Reactivity Comparison of the Design Basis and 156 IFBA Rod Assemblies at 3.0 wt% ^{235}U

a,c

Table 2-4: Reactivity Comparison of the Design Basis and 156 IFBA Rod Assemblies at 4.0 wt% ^{235}U

[ ]

a,c

Table 2-5: Reactivity Comparison of the Design Basis and 156 IFBA Rod Assemblies at 5.0 wt% ^{235}U

[ ]

[

]a,c

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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3. WCAP-17728-P, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (proprietary, not publicly available) (Enclosure 2 to letter dated March 28, 2013), Section 4.2.1, states that [[ ]]. Please explain if this is considered to be conservative because [[ ]].

Response:

[ ]a,c

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

4. WCAP-17728-P, Section 4.2.3.2, discusses axial moderator temperature profile selection, but does not discuss how PARAGON treats the moderator density. Please explain if this is the bounding moderator density profile used in the same manner as the bounding moderator temperature profile.

Response:

The axial moderator density is calculated by the FIGHTH code, as part of the core design package used at Westinghouse, which solves the steady-state heat equation, given the values of linear heat rate, burnup, flow rate, and moderator temperature. [

]^{a,c}

References:

1. WCAP-1772-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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5. To compensate for a lack of critical experiments containing fission products, [[ ]] as described in WCAP-17728-P, Section 5.3.2.1.4, and is assessed based on preliminary research performed by Oak Ridge National Laboratory (ORNL). More recent research performed by ORNL in NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions," April 2012 (ADAMS Accession No. ML12116A128), shows that 1.5 percent of the minor actinide and fission product worth (treated as a bias) is acceptable to account for the lack of a sufficient number of applicable critical experiments containing minor actinides and fission products. Applying the NUREG/CR-7109 recommendations for determining uncertainty attributed to fission product and minor actinide validation gaps, the NRC staff estimates that the licensees approach would produce a non-conservative uncertainty estimate by approximately 100 to 200 percent millirho (pcm), but the actual value could be larger. [[ ]] please provide a justification for not including the minor actinides and for not applying the results of the more recent research in the manner recommended.

Response:

[

]a,c

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a,c

Table 5-1: [ ]a,c

[

]a,c

[

]a,c

a,c

Table 5-2: [ ]a,c

[

]a,c

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.
2. ORNL/TM-12973, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," Oak Ridge National Laboratory, July 1996.

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6. For the minimum margin case in the accident analysis, which occurs with a multiple assembly misload, there is approximately 100 pcm to the regulatory k_{eff} limit of 0.95. Since the margin for the limiting accident case is minimal, please confirm that there is no increase in the total bias and uncertainty term due to not considering the presence of soluble boron when determining the combined bias and uncertainty term for this minimum margin case. The NRC staff notes that WCAP-17483-P, "Westinghouse Methodology for Spent Fuel Pool Rack Criticality Analysis," December 2011 (proprietary, not publicly available), which WCAP-17728-P is based on, recommends using a 500 pcm bias to account for any potential increase.

Response:

To confirm that the presence of soluble boron has not increased the total bias and uncertainty term, the highest worth biases and uncertainties were recalculated in the limiting multiple misload accident model. These calculations were performed with a 2400 ppm soluble boron concentration and a fresh 5.0 wt% assembly misloaded in [

]^{a,c}

Table 6-1: Recalculated Bias and Uncertainty Term for Borated Accident Condition a,c

[ ]

[

]^{a,c}

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.
2. WCAP-17483, "Westinghouse Methodology for Spent Fuel Pool and New Fuel Rack Criticality Safety Analysis," December 2011.

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7. A publication titled, "Atomic Weights of the Elements: Review 2000," from the Journal of Pure and Applied Chemistry, Volume 75, Number 6, pp. 683-800, from 2003 shows that the B-10 isotopic fraction can be as low as 0.192 in general for naturally occurring terrestrial samples, with one study showing samples with a B-10 isotopic fraction as low as 0.1893. Since the margin for the limiting accident case is minimal, please justify the B-10 isotopic fraction of [[ ]].

Response:

The boric acid used in the Reactor Coolant System and Spent Fuel Pools at CPNPP is purchased in accordance with quality assurance standards, which require that the boric acid be "undepleted in boron 10 isotope", and that the manufacturer is required to supply an isotopic analysis report of the B-10 composition.

A review of these records at CPNPP from 1992 to 2012 indicates that the boric acid utilized at CPNPP has been near the nominal value of a 0.199 atom fraction. The average value of these receipts was 0.199 atom fraction B-10, and the minimum value in this 20 year period was 0.1973.

To verify the current B-10 content in the Spent Fuel Pools, CPNPP performed B-10 isotopic analysis on water sampled from each pool on 10/31/2013. The isotopic analysis results demonstrate a B-10 content of 0.1982 atom fraction for Spent Fuel Pool 1, and 0.1977 atom fraction for Spent Fuel Pool 2. These results demonstrate that the assumptions utilized in the analysis are conservative based on current conditions.

Potential decreases in SFP B-10 concentration may occur due to three possible sources:

A. First, it is possible that Depleted Boron from the RCS may mix with the SFP water, lowering the B-10 concentration. During a normal cycle of operation, the RCS Boron concentration is diluted to very low values (<100 ppm). Prior to opening the transfer canal gates to allow core offload (therefore connecting the RCS to the SFP), the RCS boron is increased to a value greater than 2400 ppm. This addition of fresh boron typically establishes a B-10 concentration very near the natural value prior to mixing the RCS with the SFP water. However, it is possible that a mid-cycle shutdown may result in a boration from a much higher initial concentration of depleted B-10, which may result in a lower than expected B-10 concentration. This potential is addressed in the response to RAI #15, which demonstrates that when the transfer system is opened, it is not feasible that the refueling cavity water could have a B-10 concentration less than [ ][a,c].

   To ensure the impacts of any abnormal RCS depletion scenarios on the SFP are well understood in the future, CPNPP will review the calculated B-10 concentration in the RCS each refueling outage (after borating to >2400 ppm, but not including the fill of the Refueling Cavity). If the calculated value is below [ ][a,c], a B-10 measurement will be performed on the Spent Fuel Pool after adequate mixing time has occurred, but prior to the next refueling outage, to ensure the B-10 value in the SFP has not significantly changed.

B. Secondly, it is possible that the addition of fresh boric acid to the SFP, which was created from naturally occurring boron with abnormally low B-10 content, may reduce the SFP B-10 concentration. Unlike the RCS, there is rarely (if ever) a need to actively reduce or increase the SFP Boron concentration; therefore, fresh boric acid is rarely added to the Spent Fuel Pool. The Boron concentration cannot be reduced below 2400 ppm due to restrictions of Technical Specification 3.7.16 and other administrative controls. Any potential future addition of boric acid to the SFP (even at extremely low values of B-10 concentration) could only have a minor impact on B-10. For example, assume the SFP boron concentration is at 2400 ppm, with a B-10 atom fraction of 0.197. If the SFP boron is significantly increased by 200 ppm using fresh boric acid with an abnormally low atom fraction of 0.189, the resulting SFP B-10 content would be reduced to 0.1964 atom fraction. The SFP would need to be diluted and borated between 2400 ppm and

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2600 ppm a total of 6 times, using fresh boric acid at 0.189 atom fraction for all boration, to reduce the SFP B-10 concentration below [ ][a,c] It is concluded that it is not feasible for boric acid additions to invalidate the B-10 assumption utilized in the analysis.

To ensure the impacts of any abnormal RCS boration scenarios on the SFP are well understood in the future, CPNPP will review the SFP Boron Measurement history each refueling outage. If the SFP boron values have experienced any increase of more than 100 ppm, a review of B-10 values for Boric Acid purchased at CPNPP will be performed. If this review demonstrates that boric acid has been purchased which has a B-10 atom fraction below [ ][a,c], a B-10 measurement will be performed on the Spent Fuel Pool prior to the next refueling outage.

C. Lastly, depletion of the B-10 due to neutron activity within the SFP may reduce the B-10 concentration. This potential is addressed in the response to RAI #15, and it is concluded that it is not feasible for depletion in the SFP to invalidate the B-10 assumption utilized in this analysis. Therefore, no actions are necessary to address this B-10 reduction potential.

As discussed further in the response to Question #6, although the reactivity margin demonstrated in the limiting misload analysis is minimal, the limiting case was performed using extremely conservative conditions which were demonstrated to not be credible fuel accident conditions.

8. Please provide clarification for the following items related to fuel handling:

    a. The last paragraph in WCAP-17728-P, Section 5.5.5, says that the inspection cells can only ever contain one fuel assembly at a time. However in WCAP-17728-P, Section 5.5.2, it states that up to two assemblies can be placed together in the sipping equipment. This appears to be conflicting information. Please provide clarification.
    b. If two assemblies are allowed in the inspection cells, please explain the physical means that ensure that at least one assembly pitch is always maintained between assemblies.
    c. Please explain the physical means that ensure that one assembly pitch is always maintained between the inspection cell and the storage racks.
    d. There is a requirement for Region II that no fuel be placed in the interfacing row of the inspection cell. Please explain why the same requirement does not exist for Region I (CPNPP, Units 1 and 2, SFPs interface with Region I).
    e. Please explain the meaning of the last sentence in Section 5.5.2, which states, "Note that it is also acceptable to perform these tasks with the section of the assembly that is being manipulated above the storage racks."

Response to #8.a:

The fuel sipping equipment discussed in this section is not placed inside the inspection cells, and the conditions described in Section 5.5.2 are not related to the inspection cells.

The inspection cells are oversized SFP rack locations, which are designed to allow a fuel assembly to be lowered and rotated during fuel inspection activities, so that the full range of the assembly may be inspected without lowering and raising the underwater cameras. There are two inspection cell locations in SFP1, and a single inspection cell location in SFP2, as shown in Figures 3-1 and 3-2 of WCAP-17728-P Rev. 1. Fuel assembly Inspection activities may be performed outside of the storage racks as described in 5.5.2. Even when using the Inspection Cells to perform fuel inspections, the camera equipment would remain above the storage racks. If the assembly is partially lowered into the Inspection Cell during this activity (to aid in viewing the top sections of the fuel assembly), then the conditions and limitations for utilizing the Inspection Cells described in Section 5.5.5 would apply.

The Inspection Cell locations do NOT contain adequate room to place fuel sipping equipment. Section 5.5.2 of WCAP-17728-P Rev. 1 states that during fuel sipping conditions, "the fuel assemblies are separated by at least one assembly pitch via equipment design." This equipment design prevents the potential to insert this equipment into the inspection cells, since these cells are only 2 storage cells wide. Due to physical limitations of the racks, the type of equipment described in Section 5.5.2 could only be placed external to the storage racks, such as in the Wet Cask Pit or above empty storage cell locations.

Response to #8.b:

Two assemblies are not allowed into the inspection cells. Use of the inspection cells is described in more detail in the response to RAI 8.a.

Response to #8.c:

The requirement for utilizing the inspection cells is independent of the "one assembly pitch" requirement described in Section 5.5.2. The inspection cell usage is discussed in Section 5.5.5, and the restrictions are described in Section 6.3. The requirement to maintain empty locations around the Region II inspection cells also existed with the previous Criticality Safety Analysis at CPNPP, and procedural requirements currently exist to ensure storage cells adjacent to the inspection location are empty prior to use. The empty cells next to the inspection cell provides the physical barrier to assure "one assembly pitch" is maintained.

Response to #8.d:

For Region II, the restriction for inspection cell usage is described in 6.3, which states "fuel inspection cells in Region II can only be used if no fuel is stored in the adjacent fuel storage cells." Similar to the requirements for "empty cells" in WCAP-17728-P Rev. 1 Figure 5-1, "Allowable Storage Arrays", this restriction applies to adjacent Region II fuel storage cells, and does not apply to the Interface of Region I. With the adjacent storage cells vacant, the Inspection Cell reactivity is bound by Storage Array II-E, which allows for 1 out of 4 storage, surrounded by empty cells.

For the Region I Inspection Cell, there are no restrictions described in 6.3 since the reactivity of any fuel assembly placed into this storage cell is bound by normal fuel storage in Region I. Note that the Region I inspection cell is NOT simply a void in the racks where 4 cells were removed, but is an oversized cell location, including walls (with gaps to the adjacent cells) and BORAL neutron absorber panels. Therefore, there is no need to restrict adjacent storage locations in Region I, and no need to impose any special interface restrictions for the Region I / Region II interface.

Response to #8.e:

The preceding discussion describes how the reactivity of cleaning, inspection, reconstitution, and sipping activities are bound by the analysis for Array II-E. Since Array II-E applies for fuel storage inside the Region II racks, and the Region II racks do not contain neutron poison material, then it can be concluded that this analysis is bounding for a fuel assembly outside of the storage racks, when this assembly is more than 1 assembly pitch away from any other fuel assemblies.

This statement is a simple acknowledgment that although the Array II-E analysis is used to bound the activities described in this section, cleaning and fuel inspection activities are normally performed above the plane of the storage racks, or in an area which does not contain storage racks, such as the fuel transfer canal. In some cases, fuel is inserted into a storage cell, and the inspection or other activities will take place on the upper part of the fuel assembly as the fuel is raised or lowered. These activities are bound by the Array II-E storage analysis.

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9. For Array II-A, depicted in WCAP-17728-P, Section 5.2, [[

]] It was not obvious to the NRC staff that the unconsidered misload scenario for Array II-A would be non-limiting for the misload analysis. Since this misload scenario is credible, please demonstrate that a fresh assembly misload in Array II-A is non-limiting.

Response:

[

]a,c

Case 1

WALL	6	**F**	4	4
	6	6	X	4

Case 2

WALL	6	6	4	4
	6	**F**	X	4

Case 3

WALL	6	6	4	4
	F	6	X	4

Case 4

WALL	**F**	6	4	4
	6	6	X	4

Figure 9-1 Misload Cases in Array II-A

Table 9-1: [ ]a,c

a,c

[

]

Table 9-2: [ ][a,c]

a,c

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

10. Please explain if the misplacement of a fuel assembly is only possible in what is shown in WCAP-17728-P, Figures 3-1 and 3-2, as the inspection cell regions. If other regions, other than the inspection cell area, are open to the misplacement of fuel assemblies, please identify them.

Response:

It is not physically possible to place a fuel assembly between the outer boundaries of the storage racks and the SFP walls, or to misplace a fuel assembly within the outer boundaries of the storage racks in locations other than a storage cell or the inspection cells.

The CPNPP Region I cells in each Spent Fuel Pool contain a small number of 'cut-off' cells near the Spent Fuel Pool Swing Gate (these locations are identified in Figures 3-1 and 3-2 of WCAP-17728-P Rev. 1). These cells are physically identical to a normal storage cell, including placement and length of the BORAL neutron absorbers, with the exception that the top 12" of the cell (which is above the active fuel region and BORAL poison region) has been removed to prevent contact with the SFP Swing Gate as it opens into the pool. Although it is physically possible to insert fuel into these locations, fuel assemblies are restricted from these cells due to limitations of the Swing Gate.

11. There appears to be a typo in Section 5.7.4, which states, "...provides 0.05 Δk of reactivity suppression." It appears that this value should be corrected to 0.005 Δk.

Response:

The statement "...provides 0.05 Δk of reactivity suppression." in Reference 1 is correct as listed. The value 0.05 Δk refers to the approximate margin from criticality provided by 320 ppm of soluble boron. Table 5-20 of Reference 1 shows the neutron multiplication factor (k_{eff}) of each array assuming a soluble boron concentration of 320 ppm in the spent fuel pool, including biases and uncertainties but not including administrative margin. The difference between a k_{eff} of 0.995 (1.0 - 0.005 Δk administrative margin) and the k_{eff} values shown in Table 5-20 gives the reactivity worth of 320 ppm of soluble boron for each configuration. For Array II-A this value is 0.995 – 0.94474 = 0.05026 Δk.

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

12. In Region I, the rack modules are designed with a flux trap, but WCAP-17728-P does not discuss credit of the flux trap gap during a seismic event due to structural considerations. Since it is possible for the flux trap gap size to change during seismic activity, please explain why full credit of the Region I rack module flux trap gap during a seismic event is appropriate.

Response:

The CPNPP storage racks are designed to meet the seismic Category I requirements of RG 1.13 and RG 1.29. "Water Gap Flats" are part of the rack design, and are welded between the storage cells to ensure the spacing is maintained at all times, including during a seismic event.

As stated in the "No Significant Hazards Consideration" section of LAR 13-01, "the margin of safety with respect to mechanical, material or structural considerations is not changed by this proposed License Amendment Request." This structural design of the racks has been considered previously, and approved by the NRC, in Amendment 87 of the CPNPP Operating License (reference ML012560143). In the "Mechanical, Material and Structural" section of Attachment 2 to the related License Amendment Request, the following statements are made:

"The Region I / Region II racks have a sufficient margin of safety against tilting and deflection or movement during a seismic event. The Region I / Region II racks do not impact each other or the pool walls, damage fuel assemblies, or cause criticality concerns during a postulated seismic event."

"The Region I / Region II rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) were calculated under the dynamic loading conditions. All of the calculated weld stresses are less than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate."

Therefore, previously approved analysis demonstrates that the flux trap gap assumed in WCAP-17728-P Rev. 1 will remain intact during a seismic event, and crediting this gap in the criticality analysis is appropriate.

Also, note that due to the fact that the SFP Storage Racks are not restrained or physically attached to the SFP liner, it is possible the inter-rack gaps may be reduced during a seismic event. Section 5.7.4 of WCAP-17728-P Rev. 1 addresses the potential for the rack modules to slide together during a seismic event and the potential reduction in the space between the modules.

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13. Please provide details on how B-10 areal density manufacturing variation, absorber thickness variation, and degradation of BORAL absorption ability over time is accounted for in the criticality analyses. As a minimum, provide answers to the following clarification questions.

    a. Is the [[ ]] areal density the adjusted neutron absorber loading accounting only for the absorber thickness tolerance (i.e. [[ ]])?

    b. Please explain why the neutron absorber thickness tolerance is not listed in Tables 3-8 or 5-3 even though it was accounted for by adjusting the neutron absorber areal density. What is this tolerance based on?

    c. Please explain if the B_4C density bias is applied in Tables 5-8 and 5-13 based on a tolerance perturbation for B-10 areal densities of [[ ]] and [[ ]] as suggested by Table 5-3, Note 1.

    d. The values in Table 5-3 imply B-10 areal density values, but Tables 5-8 and 5-13 list a B_4C density bias. Please explain if the B_4C density values were adjusted in the KENO models to match the target B-10 areal densities given in Table 5-3. Please provide the KENO material specifications used to model Boral.

    e. Is [[ ]] the minimum certified B-10 areal density?

    f. Section 5.1.2.4, "Impact of Potential BORAL Blistering," paragraph 3 states that the areal density is adjusted from [[ ]] to [[ ]] to account for [[ ]] – this adds additional confusion as Table 5-3 does not mention [[ ]] Please explain how is this adjustment accounted for in the criticality safety analysis?

Response:

a. A tolerance on the Boral thickness was not specified by the manufacturer. [ ]^{a,c}

b. Please see response to a.

c. [ ]^{a,c}

d. [ ]^{a,c}

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[

]a,c

e. Yes, [ ]a,c is the minimum certified areal density.

f. Please see response to a.

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14. In WCAP-17728-P, Section 5.3.2.1.2, it is stated that the burnup measurement uncertainty is taken to be "the reactivity change associated with a [[ ]] change in burnup," however it is not explained why a value of [[ ]] is appropriate for this uncertainty term. Please provide justification for use of a [[ ]] change in burnup for the burnup measurement uncertainty.

Response:

Multiple industry studies have been performed to determine the accuracy of reactor burnup records. NUREG/CR-6998 (Reference 1) involved evaluation of several thousand in-core measured assembly burnup values. Reference 1 states that "utility records for fuel burnup are accurate for individual spent fuel assemblies to at least 5% of "true" assembly burnup." EPRI Report TR-112054 (Reference 2) presents an additional study based on in-core measurement comparisons, which confirms that the burnup measurement uncertainty is less than [ ]a,c. According to Section 7.2 of Reference 1 and Section 4.3 of Reference 2, the uncertainty in the utility-assigned burnup measurement values is less than the [ ]a,c value used in Reference 4 when computer models, correctly normalized to start-of-cycle conditions and adjusted periodically on the basis of in-core measurements, are used. Additionally, Westinghouse performed a study of assembly power uncertainty which is documented in Reference 3, confirming that deviation of the measured values from the reactor records is within [ ]a,c. Because Comanche Peak has used core-follow systems such as CONFORM and the BEACON™ Core Monitoring System throughout its operating history to adjust the estimated assembly burnups based on flux map results, the use of a burnup measurement uncertainty value of [ ]a,c for the discharge burnups is conservative.

References:

1. NUREG/CR-6998, "Review of Information for Spent Nuclear Fuel Burnup Confirmation," December 2009.
2. TR-112054, "Determination of the Accuracy of Utility Spent-Fuel Burnup Records," EPRI, July 1999.
3. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
4. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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15. The revised Bases for TS 3.7.16 states that "the effect of B-10 depletion on the boron concentration for maintaining k_{eff} less than or equal to 0.95 is accounted for in [[ ]]," however, B-10 depletion is not discussed in WCAP-17728-P. Please explain how is it accounted for.

Response:

[

]^{a,c}

The other potential source of depleted ^{10}B in the SFP is from mixing SFP and RCS water. The soluble ^{10}B in the RCS depletes during operation due to the absorption of neutrons. As the ^{10}B depletes, the soluble boron concentration decreases as reactor operators remove soluble boron from the RCS to maintain criticality. The ^{10}B at% is lowest (most depleted) at the end of cycle when the plant shuts down for refueling, this is also the time that the RCS contains the least amount of soluble boron. Once the plant has shut down and is in Mode 5 or 6 for refueling, the RCS is borated to at least the minimum allowable SFP soluble boron concentration (2400 ppm). The RCS boration is performed using undepleted soluble boron from the Boron Accumulator Tank (BAT).

[

]^{a,c}

References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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16. If the [[ ]] is included in the area of applicability (AOA) analysis, please explain why Boral minimum areal density is not included.

Response:

[

]a,c

17. WCAP-17728-P, Section 6.1, contains tables with various coefficients to be used with an equation relating the initial fuel enrichment to the minimum burnup for fuel assembly loading into the various storage arrays (i.e. these tables define, by curve fit, the various burnup and enrichment loading curves). The methodology for curve fitting is not explained. It is not clear if the curves are designed to pass directly through the explicit burnup and enrichment points or if they are somehow bounded. Please provide additional details on how the burnup and enrichment equations are developed based on the above considerations.

Response:

The following criteria were considered when generating the curve fits and fitting coefficients found in Reference 1, in order to ensure conservatism, while maintaining simplicity.

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References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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18. WCAP-17728-P, Section 6.2.1, states that outlier assemblies can be stored in arrays that do not require burnup credit without performing an analysis, but must be below the maximum fresh fuel enrichment. This general allowance is potentially problematic. For example, if a future fuel design incorporated higher enrichment blankets, this would allow more reactive fuel to be stored in both Region I and II without any re-analysis.

    This issue is also described in letter dated July 16, 2013 (supplemental information submitted by the licensee in response to NRC staff letter dated June 27, 2013, Item 4 (ADAMS Accession No. ML13175A225)), indicating that a new analysis would have to be performed before fuel can be loaded in the SFP if the fuel assembly in question cannot be categorized as Group F1 or F2. The response stated that only the first 5 parameters of WCAP-17728-P, Table 6-16 should be evaluated, however parameters 6 and 7 – [[ ]] – should also be evaluated as they are also based on the fuel design [[ ]] Please explain why parameters 6 and 7 have been excluded.

Response:

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Table 18-1: [ ]a,c

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References:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.

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19. It is evident that significant rodded operation is not anticipated by virtue of the imposed 0.1 GWd/MTU max rodded operation limit in the AOA analysis; additionally it is stated that rods were not inserted into the core more than 20 cm at any given time. Consequently, past and current fuel cycles appear to be reasonably covered by the as-defined AOA; however, this has not been demonstrated for future fuel cycles that do not fall within the AOA.

    WCAP-17728-P, Section 6.2.1, regarding future rodded operation not covered by the CPNPP AOA, states that [[

    ]] However, in the supplemental information provided by letter dated July 16, 2013 (Item 4), the licensee stated the following with respect to depletion parameters of future fuel assemblies that fall outside of the defined AOA:

    If the parameter only impacts the fuel depletion assumptions of WCAP-17728-P and the fuel needs to be stored in the SFP, it shall be placed in either Region I or in Array II-E in Region II.

    This statement is not consistent with the methodology presented in WCAP-17728-P, Section 6.2.1 mentioned above and specifically imposes a requirement for a burned fuel assembly outside of the AOA based on depletion characteristics to be stored in either Region I or in Region II (i.e., the 1 out of 4 storage array configuration) as if it were fresh. Based on the conflicting information above, please provide clarification for how outlier fuel assemblies that do not meet the fuel depletion criteria in the AOA defined by Table 6-16 in WCAP-17728-P will be stored for CPNPP.

Response:

<u>CPNPP approach for treating HFP rodded operation assumption outliers:</u>

The CPNPP response to Question #4 in the supplemental information was accurate at the time, since CPNPP originally planned to treat all 'outliers', or fuel assemblies which do not satisfy the depletion assumptions of WCAP-17728-P Rev. 1 Table 6-16, the same. This included outliers which did not satisfy the assumption for Maximum HFP Rodded Operation. CPNPP was not, at the time, planning on utilizing the provision in the analysis which states "assemblies which are classified as outlier assemblies because of HFP Rodded Operation can be stored using burnup credit if the burnup accrued during rodded operation is not credited."

The decision to not utilize this provision was made based on CPNPP's understanding at that time that outliers to the AOA (Area of Applicability) would be evaluated per 50.59, and evaluations for outliers would not require NRC approval (assuming the supporting analysis demonstrated that the resulting reactivity impacts were not adverse). Based on clarifying discussions both internally and with the NRC, CPNPP now understands the AOA is considered part of the supporting methodology; therefore the required evaluation of AOA outliers will require NRC approval of the supporting analysis.

Based on this revised understanding, CPNPP plans on utilizing the provision in WCAP-17728-P Rev. 1 for applying burnup credit to store fuel assemblies which are classified as outlier assemblies solely due to HFP Rodded Operation. For these assemblies, burnup which is accrued during HFP rodded conditions will NOT be credited in the Technical Specification Surveillance, but all other burnup accrued during the cycle will be credited.

For all other depletion parameters in Table 6-16 of WCAP-17728-P Rev.1 (Including HFP Rodded Operation parameter in combination with another depletion parameter), CPNPP will not apply any credit for burnup in the Technical Specification Surveillance, unless prior approval is obtained from the NRC.

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Justification for not crediting burnup accrued during HFP rodded operation:

As implied in Section 6.2.1 of WCAP-17728-P Rev.1, the depletion which occurs under rodded conditions may be ignored to ensure that the reactivity assumed in the Technical Specification surveillance is bounded by the analysis. Section 5.8 of WCAP-17728-P Rev.1 describes how the reactivity of an assembly experiencing rodded operation can increase relative to an assembly which does not experience rodded operation, due to several factors. Due to this potential, the depletion which occurs under these conditions cannot be credited in the surveillance without further analysis. This comparison between rodded and unrodded operation describes the relative reactivity between two fuel assemblies with the same burnup value, and does not imply that the fuel assembly reactivity, at any axial location, could actually increase due to depletion under rodded conditions. In other words, rodded depletion, even under the most extreme rodded conditions, cannot increase the reactivity at any axial location in the fuel assembly. Therefore, it is inherently conservative to ignore this depletion, and only credit the burnup which accrued under normal operating conditions.

For example, assume that a Fuel Assembly is depleted for 200 days at normal HFP ARO conditions. This assembly then experiences 100 days of HFP rodded conditions. The TS 3.7.17 surveillance for the fuel assembly would only credit 200 EFPD (Effective Full Power Days) of burnup, since the 100 days of depletion during rodded operation is not credited. Due to the additional uncredited depletion time, the assembly is inherently less reactive than it was following the 200 days of normal HFP ARO operation; specifically it will be less reactive at each axial location (including areas which were covered by control rods, which experienced some amount of fuel depletion even under these abnormal conditions), regardless of the final axial burnup profile.

In a more realistic scenario, the fuel will likely experience additional unrodded depletion following the period of rodded operation. Section 5.8 of WCAP-17728-P Rev. 1 describes that "once the RCCA has been withdrawn from the assembly, power preferentially moves to the under-depleted top of the assembly and over time the axial burnup profile developed will return to a profile typical of unrodded operation." If 100 days of rodded operation for the fuel assembly in the example above had occurred prior to, or during, the 200 days of HFP ARO operation, the axial burnup profile would eventually return to a profile similar to a fuel assembly depleted under unrodded conditions, due to the preferential depletion of the under-depleted axial locations. In this case, the reactivity assumed by crediting only 200 EFPD of burnup in the surveillance bounds the actual reactivity at all axial locations.

Note that the NRC has reviewed and approved a similar treatment of Rodded Operation in a recent Safety Evaluation related to SFP criticality analysis. The proposed treatment of rodded operation burnup is similar to the commitments documented in "PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: SPENT FUEL POOL CRITICALITY CHANGES (TAC NOS. ME6984 AND ME6985)," dated August 29, 2013, Reference ML13241A383. On page 9 of the supporting Safety Evaluation Report (Enclosure 3 to this document), the following Prairie Island commitment is described:

> *"In conjunction with implementation of the amendment, procedures will be revised to require an assessment of a fuel assembly's exposure to rodded power operation in the core prior to moving that fuel assembly into the spent fuel pool (SFP) storage racks. If an assembly experiences more than 100 megawatt day per metric ton uranium (MWd/MTU) of core average full-power rodded operation exposure, this exposure experienced while rodded will not be credited for determining the coefficients used to categorize fuel assemblies as described in WCAP-17400-P."*

In the discussion following the commitment, the Safety Evaluation concluded that the approach was acceptable:

> *"The current version of the commitment does not allow for significant amounts of rodded operation to be considered when determining whether a fuel assembly meets the storage requirements. Therefore, the NRC concludes that this commitment acceptably accounts for the variability of rodded operation."*

Treatment of HFP rodded operation outliers in past cycles:

Section 5.8 of WCAP-17728-P Rev.1 discusses the potential impacts of Rodded Operation on fuel reactivity, and includes a statement that "Comanche Peak has not operated at full power with control rods inserted a significant length... Therefore, there is no significant burnup accrued during depletion with RCCAs inserted in the active fuel height, and no need to account for these effects in burnup limits contained within this analysis."

Section 6.2 summarizes the Area of Applicability of the analysis, which includes key assumptions utilized in the analysis which will be "confirmed for each cycle of operation to assure that the results presented here remain valid." The Maximum HFP Rodded Operation in the AOA is 0.1 GWD/MTU/cycle.

CPNPP has recently completed a more detailed review of past plant history to confirm the AOA assumptions related to Maximum HFP Rodded Operation. Although the vast majority of past CPNPP operation was performed under unrodded conditions, and the general statements made in Section 5.8 regarding past operation are valid, there are several past cycles which do not satisfy the conservative threshold of 0.1 GWD/MTU/cycle of full power operation below 210 steps (from Table 6.16).

Because the depletion analysis did not specifically address the time spent during HFP Rodded Operation, CPNPP plans on treating rodded fuel assemblies from past cycles in a manner identical to future cycles, i.e., the fuel depletion which occurred during HFP Rodded Operation will NOT be credited in the Technical Specification Surveillance for any cycle which accrued more than 0.1 GWD/MTU of HFP Rodded Operation.

The administrative controls and Configuration Confirmation Software tools described in Enclosure 1 of LAR 13-01 will incorporate limitations to ensure that the appropriate burnup is credited for fuel assemblies which have experienced HFP Rodded Operation beyond the low threshold required by the AOA. As described in Enclosure 1 of LAR 13-01, only use of QA-controlled software is permitted for performance of SR 3.7.17.1.

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20. Please explain why the axial burnup profile evaluated in the AOA is not defined by Table 6-16.

Response:

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Reference:

1. WCAP-17728-P, Revision 1, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," October 2013.