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

July 1, 2014

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

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATIONS 3.7.16, "FUEL STORAGE POOL BORON CONCENTRATION," 3.7.17, "SPENT FUEL ASSEMBLY STORAGE," 4.3, "FUEL STORAGE," AND 5.5, "PROGRAMS AND MANUALS" (TAC NOS. MF1365 AND MF1366)

Dear Mr. Flores:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 162 to Facility Operating License No. NPF-87 and Amendment No. 162 to Facility Operating License No. NPF-89 for Comanche Peak Nuclear Power Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 28, 2013, as supplemented by letters dated July 16, October 22, November 26, and December 17, 2013, and January 16, April 17, and May 1, 2014.

The amendments revise TS 3.7.16, "Fuel Storage Pool Boron Concentration," TS 3.7.17, "Spent Fuel Assembly Storage," TS 4.3, "Fuel Storage," and TS 5.5, "Programs and Manuals," for storage of uprated fuel in Region II of the spent fuel pool. TS 3.7.16 describes the minimum concentration of dissolved boron in the fuel storage pools. TS 3.7.17 describes the storage configurations allowed in Region II high density storage racks based on minimum bunup limitations generated from a spent fuel pool criticality analysis. TS 4.3 describes the fuel storage design requirements in the Fuel Building. TS 5.5 provides a Neutron Absorber Monitoring Program.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public inspections, exemptions, requests for withholding." Proprietary information is indicated by text enclosed within brackets. Accordingly, the NRC staff has also prepared a non-proprietary publically available version of the SE, which is provided in Enclosure 3. The proprietary version of the SE is provided in Enclosure 4.

NOTICE: Enclosure 4 to this letter contains Proprietary Information. Upon separation from Enclosure 4, this letter is DECONTROLLED.

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Balwant K. Singal

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 162 to NPF-87
2. Amendment No. 162 to NPF-89
3. Safety Evaluation (non-proprietary)
4. Safety Evaluation (proprietary)

cc w/encls 1, 2, and 3: Distribution via Listserv

ENCLOSURE 1

AMENDMENT NO. 162

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-445

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

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. NPF-87

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Luminant Generation Company LLC dated March 28, 2013, as supplemented by letters dated July 16, October 22, November 26, and December 17, 2013, and January 16, April 17, and May 1, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 162 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-87 and Technical Specifications

Date of Issuance: July 1, 2014

ENCLOSURE 2

AMENDMENT NO. 162

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-446

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

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. NPF-89

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Luminant Generation Company LLC dated March 28, 2013, as supplemented by letters dated July 16, October 22, November 26, and December 17, 2013, and January 16, April 17, and May 1, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-89 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 162 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-89 and
Technical Specifications

Date of Issuance: July 1, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 162

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 162

TO FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Facility Operating License Nos. NPF-87 and NPF-89, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-87

REMOVE	INSERT
3	3

Facility Operating License No. NPF-89

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.7-36	3.7-36
3.7-37	3.7-37
3.7-38	3.7-38
3.7-39	3.7-39
3.7-40	3.7-40
3.7-41	3.7-41
3.7-42	3.7-42
3.7-43	3.7-43
3.7-44	3.7-44
3.7-45	3.7-45
3.7-46	3.7-46
3.7-47	3.7-47
3.7-48	3.7-48
---	3.7-49
4.0-2	4.0-2
4.0-3	4.0-3
5.5-18	5.5-18
---	5.5-19

(3) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Luminant Generation Company LLC is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal through Cycle 13 and 3612 megawatts thermal starting with Cycle 14 in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 162 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Luminant Generation Company LLC is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal through Cycle 11 and 3612 megawatts thermal starting with Cycle 12 in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 162 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DELETED

3.7 PLANT SYSTEMS

3.7.16 Fuel Storage Pool Boron Concentration

LCO 3.7.16 The fuel storage pool boron concentration shall be ≥ 2400 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool	Immediately
	AND	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the fuel storage pool boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program.

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 New or spent fuel assemblies will be stored in compliance with Figure 3.7.17-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II racks of the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Initiate action to move fuel as necessary to restore compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the acceptability of fuel movement plans and the resulting storage configuration in accordance with Figure 3.7.17-1.	Prior to moving a fuel assembly into any Region II storage location.

Figure 3.7.17-1 (page 1 of 3)
Spent Fuel Pool Loading Restrictions

All 2x2 Region II storage cell arrays shall comply with one of the Arrays Definitions below. Each storage location is a corner location for up to 4 separate 2x2 arrays.

A. Arrays II-A through II-E designate the pattern of fuel which may be stored in any 2x2 Array, and are dependent upon Fuel Category.

B. Fuel Categories 1-6 are determined based on Fuel Burnup, Initial Enrichment, Decay Time, and Fuel Group.

C. Fuel Group F1 assemblies have a nominal rod outer diameter of 0.374 inches. Fuel Group F2 assemblies have a nominal rod outer diameter of 0.360 inches.

Array Definition	Illustration			
Array II-A Category 6 assembly in every cell. Only valid for two rows adjacent to the SFP wall. The two rows adjacent to Array II-A must be Array II-B, and the empty cell in Array II-B must be adjacent to Array II-A.	W A L L	6 6	6 6	Array II-B
Array II-B Cateory 4 fuel assembly in 3 out of 4 cells, with empty cell in the fourth cell.		4 X	4 4	
Array II-C Pattern which contains fuel in 3 out of 4 cells, including two diagonally-opposed Category 5 assemblies, one Category 3 assembly, and one empty location. Only Fuel Group F2 assemblies may be stored in Array II-C.		5 X	3 5	
Array II-D Checkerboard pattern of two diagonally-opposed Category 2 assemblies with two diagonally-opposed empty cells.		2 X	X 2	
Array II-E 1 out of 4 storage array, with 3 empty cells.		X X	X 1	

Figure 3.7.17-1 (page 2 of 3)

Notes:

1. Fuel Categories are ranked in order of relative reactivity, from Category 1 to 6. Fuel Category 1 assemblies have the highest reactivity, and Fuel Category 6 assemblies have the lowest.

2. All Fresh Fuel Assemblies (assemblies with a burnup value of 0.0 MWD/MTU) should be considered Category 1 fuel, independent of Fuel Group or Enrichment.

   a. In Fuel Group F1, Fuel Category 1 is fresh fuel up to 3.5 weight percent U-235 Initial Enrichment.

   b. In Fuel Group F2, Fuel Category 1 is fresh fuel up to 5.0 weight percent U-235 Initial Enrichment.

3. Fuel Category 2 is any Non-Fresh fuel assembly up to 3.5 weight percent U-235 Initial Enrichment (Burnup Requirement is > 0 MWD/MTU).

4. For all other fuel, Fuel Categories are determined as follows:

   a. For Initial Enrichment values below the Minimum Applicable Initial Enrichment values of Table 3.7.17-1, the Fitting Coefficients of Tables 3.7.17-2 and 3.7.17-3 are not applicable. The Minimum Burnup Requirement for the corresponding Category is > 0 MWD/MTU.

   b. For Fuel Group F1 assemblies, determine the Fitting Coefficients A_1 - A_4 using Table 3.7.17-2. Note that Table 3.7.17-2 is only applicable to fuel with $\geq$ 10 years of decay time, and an Initial Enrichment of $\leq$ 3.5 weight percent.

   c. For Fuel Group F2 assemblies, determine the Fitting Coefficients A_1 - A_4 using Table 3.7.17-3.

   d. The required Minimum Burnup value (in MWD/MTU) for each Fuel Category is calculated based on Initial Enrichment (En) and the appropriate fitting coefficients, using the equation below. If the fuel assembly burnup is greater than or equal to the calculated Minimum Burnup value, then the fuel may be classified into this Fuel Category.

      $$\text{Minimum Burnup} = 1{,}000 \times [A_1 \times En^3 + A_2 \times En^2 + A_3 \times En + A_4]$$

   e. All relevant uncertainties are explicitly included in the criticality analysis. No additional allowance for burnup uncertainty or initial enrichment uncertainty is required.

Figure 3.7.17-1 (page 3 of 3)

Notes (continued):

f. Conservatively low values of Decay Time may be used to calculate the Minimum Burnup value, or interpolation may be used. If interpolation is used, Minimum Burnup values for tabulated Decay Time values above and below the actual value should first be determined. Next, linear interpolation between these values may be used to determine the Minimum Burnup value. No extrapolation beyond 20 years is permitted.

g. Initial Enrichment (En) is the nominal U-235 weight percent enrichment of the central zone region of fuel, excluding axial blankets, prior to fuel depletion.

h. If the computed Minimum Burnup value $\leq$ 0 MWD/MTU, the Minimum Burnup Requirement is > 0 MWD/MTU.

5. In all Arrays, an assembly with a higher Fuel Category number can be utilized in place of any fuel assembly with a lower Fuel Category Number, with the following exception.

   a. Fuel Group F1 assemblies are not allowed to be stored in Array II-C, regardless of Fuel Category.

6. An empty (water-filled) cell can be substituted for any fuel-containing cell in all storage arrays.

7. Any storage array location designated for a fuel assembly can be replaced with non-fissile hardware. Items other than Fuel Assemblies which contain fissile material are restricted to storage in Region I.

8. Fuel assembly inserts approved for use in the reactor core can be inserted in a stored assembly (in the Spent Fuel Pool) without affecting the fuel category.

Table 3.7.17-1

Minimum Applicable Initial Enrichment for
Table 3.7.17-2 and Table 3.7.17-3 Fitting Coefficients

FUEL CATEGORY	FUEL GROUP F1	FUEL GROUP F2
6	1.25	1.20
5	N/A	1.30
4	1.35	1.45
3	N/A	1.45
2	N/A	3.55

Table 3.7.17-2

Fuel Group F1
Nominal Fuel Rod Outer Diameter of 0.374"

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu)
as a Function of Decay Time and Initial Enrichment (En)

FUEL CATEGORY	DECAY (YRS)	FITTING COEFFICIENTS			
		A_1	A_2	A_3	A_4
6	10	1.4351	-17.3247	73.3805	-67.4585
6	15	1.7078	-18.7916	74.6322	-67.2637
6	20	0.5289	-9.9969	53.7741	-52.6302
4	10	-0.0444	-1.3474	22.7039	-28.0852
4	15	0.2015	-2.6257	24.1016	-28.2473
4	20	0.4646	-4.1432	26.3891	-29.2170

Note: Fuel must have at least 10 years of decay time and less than or equal to 3.5 weight percent Initial Enrichment to utilize Table 3.7.17-2

Table 3.7.17-3

Fuel Group F2
Nominal Fuel Rod Outer Diameter of 0.360"

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu)
as a Function of Decay Time and Initial Enrichment (En)

FUEL CATEGORY	DECAY (YRS)	FITTING COEFFICIENTS			
		A_1	A_2	A_3	A_4
6	0	0.5789	-7.4498	42.4056	-41.1591
6	5	0.5247	-6.8992	39.7676	-38.6927
6	10	0.2701	-4.4306	31.9841	-32.4674
6	15	0.3105	-4.5582	31.1825	-31.3916
6	20	0.2374	-3.8754	28.8900	-29.4975
5	0	0.9373	-11.2553	54.7226	-54.1769
5	5	0.6169	-8.1494	44.7801	-45.7968
5	10	0.5380	-7.1852	40.7044	-41.9545
5	15	0.5385	-7.0180	39.2299	-40.3213
5	20	0.5200	-6.7906	38.0244	-39.0979
4	0	0.2553	-3.9826	30.6152	-36.7967
4	5	0.2366	-3.6430	28.2160	-33.9749
4	10	0.4387	-5.6018	33.3609	-37.9327
4	15	0.5450	-6.6302	36.0760	-40.0315
4	20	0.6327	-7.4663	38.2724	-41.7257
3	0	0.5317	-6.1006	32.7118	-36.2263
3	5	0.5228	-5.9434	31.2846	-34.4602
3	10	0.5432	-6.1075	31.1578	-33.9933
3	15	0.5206	-5.8897	30.1768	-32.9600
3	20	0.5158	-5.7796	29.4050	-32.0577
2	0	0.0000	1.6738	-8.5396	9.2206

3.7 PLANT SYSTEMS

3.7.18 Secondary Specific Activity

LCO 3.7.18 The specific activity of the secondary coolant shall be ≤ 0.10 µCi/gm DOSE EQUIVALENT I-131

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify the specific activity of the secondary coolant is ≤ 0.10 µCi/gm DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program.

3.7 PLANT SYSTEMS

3.7.19 Safety Chilled Water

LCO 3.7.19 Two safety chilled water trains shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safety chilled water train inoperable.	A.1 Restore safety chilled water train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.19.1	----------NOTE---------- Isolation of safety chilled water flow to individual components does not render the safety chilled water system inoperable. ---------- Verify each safety chilled water manual, power operated, and automatic valve servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.19.2	Verify each safety chilled water pump and chiller starts on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.

3.7 PLANT SYSTEMS

3.7.20 UPS HVAC System

LCO 3.7.20 Two UPS HVAC System Trains shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One UPS HVAC System train inoperable.	A.1 Verify the affected UPS & Distribution Room is supported by an OPERABLE UPS A/C Train.	Immediately
	AND	
	A.2 Restore the inoperable UPS HVAC train to OPERABLE status.	30 days
B. Two UPS HVAC System trains inoperable.	B.1 Verify air circulation is maintained by at least one UPS A/C Train.	Immediately
OR	AND	
Required Action A.1 and associated Completion Time not met.	B.2 Verify the air temperature in the affected UPS & Distribution Room(s) does not exceed the maximum temperature limit for the room(s).	12 hours AND Once per 12 hours thereafter
	AND	
	B.3 Restore UPS HVAC System train to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action B.1 and associated Completion Time not met.	C.1 Restore the required support.	1 hour
D. Required Action and associated Completion Time of Required Action A.2, B.2, B.3 or C.1 not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.20.1	Verify each required UPS & Distribution Room Fan Coil Unit operates ≥ 1 continuous hour.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.20.2	Verify each required UPS A/C train operates for ≥ 1 continuous hour.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.20.3	Verify each required UPS A/C train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program.

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 400 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. A nominal 9 inch center to center distance between fuel storage locations in Region II fuel storage racks;

e. A nominal 10.65 inch by nominal 11.05 inch center to center distance between fuel assemblies placed in Region I fuel storage racks;

f. New or partially spent fuel assemblies may be allowed restricted storage in a 1 out of 4 configuration in Region II fuel storage racks (as shown in Figure 3.7.17-1, Array II-E) or unrestricted storage in Region I fuel storage racks;

g. Storage of new or spent fuel assemblies in Region II storage racks must comply with 3.7.17 Spent Fuel Assembly Storage.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 854 ft.

4.3.3 Capacity

The spent fuel storage pools are designed and shall be maintained with a storage capacity limited to no more than 3373 fuel assemblies.

5.5 Programs and Manuals

5.5.21 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI-04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.5.22 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

The Region I storage cells in the CPNPP Spent Fuel Pool utilize the neutron absorbing material BORAL, which is credited in the Safety Analysis to ensure the limitations of Technical Specification 4.3.1.1 are maintained.

In order to ensure the reliability of the Neutron Poison material, a monitoring program is required to routinely confirm that the assumptions utilized in the criticality analysis remain valid and bounding. The Neutron Absorber Monitoring Program is established to monitor the integrity of neutron absorber test coupons periodically as described below.

A test coupon "tree" shall be maintained in each SFP. Each coupon tree originally contained 8 neutron absorber surveillance coupons. Detailed measurements were taken on each of these 16 coupons prior to installation, including weight, length, width, thickness at several measurement locations, and B-10 content (g/cm^2). These coupons shall be maintained in the SFP to ensure they are exposed to the same environmental conditions as the neutron absorbers installed in the Region I storage cells, until they are removed for analysis.

One test coupon from each SFP shall be periodically removed and analyzed for potential degradation, per the following schedule. The schedule is established to ensure adequate coupons are available for the planned life of the storage racks.

5.5 Programs and Manuals

5.5.22 Spent Fuel Storage Rack Neutron Absorber Monitoring Program (continued)

Year	Coupon Number	Year	Coupon Number
2013	1	2028	5
2015	2	2033	6
2018	3	2043	7
2023	4	2053	8

Further evaluation of the absorber materials, including an investigation into the degradation and potential impacts on the Criticality Safety Analysis, is required if:

- A decrease of more than 5% in B-10 content from the initial value is observed in any test coupon as determined by neutron attenuation.
- An increase in thickness at any point is greater than 25% of the initial thickness at that point.

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 162 TO

FACILITY OPERATING LICENSE NO. NPF-89

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within double brackets.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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1.0 INTRODUCTION

By application dated March 28, 2013 (Reference 1), as supplemented by letters dated July 16, October 22, November 26, and December 17, 2013, and January 16, April 17, and May 1, 2014 (References 2, 3, 4, 5, 6, 7, and 8, respectively), Luminant Generation Company LLC (Luminant, the licensee) requested changes to the Technical Specifications (TSs) for Comanche Peak Nuclear Power Plant, Units 1 and 2 (CPNPP). Portions of the letters dated March 28, July 16, October 22, November 26, and December 17, 2013, and April 17, 2014, contain sensitive unclassified non-safeguards information (proprietary) and, accordingly, have been withheld from public disclosure.

The proposed changes would revise TS 3.7.16, "Fuel Storage Pool Boron Concentration," TS 3.7.17, "Spent Fuel Assembly Storage," TS 4.3, "Fuel Storage," and TS 5.5, "Programs and Manuals," for storage of uprated fuel in Region II of the spent fuel pool (SFP). TS 3.7.16 describes the proposed minimum concentration of dissolved boron in the fuel storage pools. TS 3.7.17 describes the proposed storage configurations allowed in Region II high density storage racks based on minimum burnup limitations generated from an SFP criticality analysis. TS 4.3 describes the proposed fuel storage design requirements in the Fuel Building. TS 5.5 provides a proposed Neutron Absorber Monitoring Program.

The U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination was based on letters dated March 28 and July 16, 2013. The supplemental letters dated October 22, November 26, and December 17, 2013, and January 16, April 17, and May 1, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 5, 2013 (78 FR 66391).

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954, as amended (AEA) requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The NRC regulatory requirements related to the content of the TS are contained Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications." The regulation in 10 CFR 50.36 requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The regulation requires, in part, that the TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(2)(ii), referring to requirements for limiting conditions for operation, state:

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> (A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The regulations in 10 CFR 50.36(c)(4), "Design features," state:

> 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.

The regulations in 10 CFR 50.36(c)(5), "Administrative controls," state:

> Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each

> licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4.

The regulations in 10 CFR 50.68, "Criticality accident requirements," requires the holder of an operating license for a nuclear power reactor to comply with the requirements of 10 CFR 50.68(b). Since changes are only being requested for fresh and spent fuel storage requirements in the SFP racks, the requirements of 10 CFR 50.68(b)(1) and 10 CFR 50.68(b)(4) apply and have been considered in this evaluation.

The regulations in 10 CFR 50.68(b)(1) state:

> 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.

The regulations in 10 CFR 50.68(b)(4) state:

> 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 regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 62, "Prevention of criticality in fuel storage and handling," state:

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

The NRC staff's review was performed consistent with Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" (Reference 9), and Section 9.1.2, "New and Spent Fuel Storage," subsection I.11 area of review related to chemical engineering issues (Reference 10) of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP).

On September 29, 2011, the NRC staff issued Interim Staff Guidance (ISG) DSS-ISG-2010-01 (Reference 11). The purpose of the ISG is to provide updated review guidance to the NRC staff to address the increased complexity of recent SFP nuclear criticality safety (NCS) analyses and operations. The NRC staff also used ISG DSS-ISG-2010-01 for the review of the current application.

On August 19, 1998, the NRC staff issued an internal memorandum containing guidance for reviewing criticality analyses of fuel storage at light-water-reactor (LWR) power plants. This memorandum is known colloquially as the "Kopp Letter" (Reference 12), after the author,

Laurence Kopp. While the Kopp Letter does not specify a methodology, it does provide some guidance on the more salient aspects of an NCS analysis, including computer code validation. The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), and to borated and unborated conditions. The NRC staff also used the Kopp Letter for the review of the current application.

The NRC staff used the following regulations and NRC staff guidance documents in reviewing the human factor engineering aspects of the current application:

- 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," paragraph (b) states, in part, that

  > [E]ach holder of an operating license shall establish, implement, and maintain a training program that meets the requirements of paragraphs b(2) and b(3) of this section.

- SRP Chapter 18, "Human Factors Engineering," Revision 2 (Reference 13).

- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, September 2012 (Reference 14).

- NUREG-1764, "Guidance for the Review of Changes to Human Actions," February 2004 (Reference 41).

3.0 TECHNICAL EVALUATION

3.1 Background

Luminant submitted a license amendment request (LAR) for increasing the rated thermal power of CPNPP from 3458 Megawatt thermal (MWt) to 3612 MWt by letter dated August 28, 2007 (Reference 15). The application also included a request for approval of the revised SFP criticality analyses (SFPCA) for storage of the uprated fuel in Region II of the SFP. The NRC staff approved the LAR for increasing the rated thermal power by letter dated June 27, 2008 (Reference 16), but the licensee withdrew the request for approval of the revised SFPCA due to significant issues identified by the NRC staff. However, later the licensee started storing the uprated fuel in Region II of the SFP by use of NRC Administrative Letter 98-10 (Reference 17) by establishing additional administrative controls and did not seek NRC staff approval for this action. By Confirmatory Action Letter (CAL) dated October 22, 2012 (Reference 18), the NRC staff outlined procedural administrative controls to remain in place until a license amendment could be approved by the NRC staff, including a restriction on any fuel movements in Region II of the SFP. By letter dated January 15, 2013 (Reference 19), the licensee submitted an LAR to address the interim condition to comply with Commitment 3 of the CAL. However, later this request was withdrawn by the licensee in the wake of the LAR submitted by letter dated March 28, 2013 (Reference 1), to comply with Commitment 1 of the CAL.

The licensee has proposed changes to TS 3.7.16, "Fuel Storage Pool Boron Concentration," TS 3.7.17, "Spent Fuel Assembly Storage," TS 4.3, "Fuel Storage," and TS 5.5, "Programs and Manuals." The proposed changes to TS 3.7.16 reflect a change in the required fuel storage

pool soluble boron concentration based on the results of a new criticality analysis provided in WCAP-17728-P, "Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Reference 20). The proposed changes to TS 3.7.17 include new SFP loading restrictions in terms of allowable storage patterns, and minimum burnup requirements as a function of enrichment, fuel type, and fuel reactivity category; these changes are also supported by WCAP-17728-P. The proposed changes to TS 4.3 includes updates to the minimum soluble boron concentration, Region I fuel assembly spacing, specific new or partially spent fuel assembly storage restrictions in Region II consistent with TS 3.7.17, and general Region II storage restrictions consistent with TS 3.7.17. The proposed change to TS 5.5, "Programs and Manuals," adds TS program 5.5.22, "Neutron Absorber Monitoring Program."

The new neutron absorber monitoring program proposed to be added as TS program 5.5.22 is intended to ensure that the criticality analysis assumptions related to the Boral absorber material will remain applicable throughout the lifetime of the plant. The licensee has also proposed to implement augmented administrative controls to reduce the potential for a multiple fuel assembly misloading event.

3.2 Proposed TS Changes

TS 3.7.16, Fuel Storage Pool Boron Concentration

Current LCO 3.7.16 states:

> The fuel storage pool boron concentration shall be ≥ 2000 ppm.

Revised LCO 3.7.16 would state:

> The fuel storage pool boron concentration shall be ≥ 2400 ppm.

TS 3.7.17, Spent Fuel Assembly Storage:

Current LCO 3.7.17 states:

> The combination of initial enrichment, burnup and decay time of each spent fuel assembly stored in Region II racks shall be within either (1) the "acceptable" domain of Figure 3.7.17-1 in a 4 out of 4 configuration, (2) the "acceptable" domain of Figure 3.7.17-2 in a 3 out of 4 configuration, (3) the "acceptable" domain of Figure 3.7.17-3 in a 2 out of 4 configuration, or (4) shall be stored in a 1 out of 4 configuration. The acceptable storage configurations are shown in Figure 3.7.17-4.

Revised LCO 3.7.17 would state:

> New or spent fuel assemblies will be stored in compliance with Figure 3.7.17-1.

Current APPLICABILITY states:

> Whenever any fuel assembly is stored in Region II racks of the spent fuel storage pool.

Revised APPLICABILITY would state:

> Whenever any fuel assembly is stored in Region II of the spent fuel storage pool.

Current Required Action A.1 states:

> Initiate action to move the noncomplying fuel assembly to an acceptable storage location.

Revised Required Action A.1 would state:

> Initiate action to move fuel as necessary to restore compliance.

Current SR 3.7.17.1 states:

> Verify by administrative means the initial enrichment, burnup and decay time of the fuel assembly is in accordance with either (1) the "acceptable" domain of Figure 3.7.17-1 in a 4 out of 4 configuration, (2) the "acceptable" domain of Figure 3.7.17-2 in a 3 out of 4 configuration, (3) the "acceptable" domain of Figure 3.7.17-3 in a 2 out of 4 configuration, or (4) a 1 out of 4 configuration. The acceptable storage configurations are shown in Figure 3.7.17-4.

Revised SR 3.7.17.1 would state:

> Verify by administrative means the acceptability of fuel movement plans and the resulting storage configuration in accordance with Figure 3.7.17-1.

The current frequency for SR 3.7.17.1 states

> Prior to storing the fuel assembly in Region II racks

The revised frequency for SR 3.7.17.1 would state:

> Prior to moving a fuel assembly into any Region II storage location.

The licensee proposes to delete current Figures 3.7.17-1 (page 1 of 1), 3.7.17-2 (page 1 of 1), 3.7.17-3 (page 1 of 1), and 3.7.17-4 (page 1 of 1) and replace them with Figure 3.7.17-1 (page 1 of 3, page 2 of 3, and page 3 of 3) and Tables 3.7.17-1, 3.7.17-2, and 3.7.17-3, as shown below:

Figure 3.7.17-1 (page 1 of 3)

Spent Fuel Pool Loading Restrictions

All 2x2 Region II storage cell arrays shall comply with one of the Arrays Definitions below. Each storage location is a corner location for up to 4 separate 2x2 arrays.

A. Arrays II-A through II-E designate the pattern of fuel which may be stored in any 2x2 Array, and are dependent upon Fuel Category.

B. Fuel Categories 1-6 are determined based on Fuel Burnup, Initial Enrichment, Decay Time, and Fuel Group.

C. Fuel Group F1 assemblies have a nominal rod outer diameter of 0.374 inches. Fuel Group F2 assemblies have a nominal rod outer diameter of 0.360 inches.

Array Definition	Illustration			
Array II-A Category 6 assembly in every cell. Only valid for two rows adjacent to the SFP wall. The two rows adjacent to Array II-A must be Array II-B, and the empty cell in Array II-B must be adjacent to Array II-A.	W A L L	6	6	Array II-B
		6	6	

Array II-B Cateory 4 fuel assembly in 3 out of 4 cells, with empty cell in the fourth cell.	4	4
	X	4

Array II-C Pattern which contains fuel in 3 out of 4 cells, including two diagonally-opposed Category 5 assemblies, one Category 3 assembly, and one empty location. Only Fuel Group F2 assemblies may be stored in Array II-C.	5	3
	X	5

Array II-D Checkerboard pattern of two diagonally-opposed Category 2 assemblies with two diagonally-opposed empty cells.	2	X
	X	2

Array II-E 1 out of 4 storage array, with 3 empty cells.	X	X
	X	1

Figure 3.7.17-1 (page 2 of 3)

Notes:

1. Fuel Categories are ranked in order of relative reactivity, from Category 1 to 6. Fuel Category 1 assemblies have the highest reactivity, and Fuel Category 6 assemblies have the lowest.

2. All Fresh Fuel Assemblies (assemblies with a burnup value of 0.0 MWD/MTU) should be considered Category 1 fuel, independent of Fuel Group or Enrichment.

   a. In Fuel Group F1, Fuel Category 1 is fresh fuel up to 3.5 weight percent U-235 Initial Enrichment.

   b. In Fuel Group F2, Fuel Category 1 is fresh fuel up to 5.0 weight percent U-235 Initial Enrichment.

3. Fuel Category 2 is any Non-Fresh fuel assembly up to 3.5 weight percent U-235 Initial Enrichment (Burnup Requirement is > 0 MWD/MTU).

4. For all other fuel, Fuel Categories are determined as follows:

   a. For Initial Enrichment values below the Minimum Applicable Initial Enrichment values of Table 3.7.17-1, the Fitting Coefficients of Tables 3.7.17-2 and 3.7.17-3 are not applicable. The Minimum Burnup Requirement for the corresponding Category is > 0 MWD/MTU.

   b. For Fuel Group F1 assemblies, determine the Fitting Coefficients A_1 - A_4 using Table 3.7.17-2. Note that Table 3.7.17-2 is only applicable to fuel with $\geq$ 10 years of decay time, and an Initial Enrichment of $\leq$ 3.5 weight percent.

   c. For Fuel Group F2 assemblies, determine the Fitting Coefficients A_1 - A_4 using Table 3.7.17-3.

   d. The required Minimum Burnup value (in MWD/MTU) for each Fuel Category is calculated based on Initial Enrichment (En) and the appropriate fitting coefficients, using the equation below. If the fuel assembly burnup is greater than or equal to the calculated Minimum Burnup value, then the fuel may be classified into this Fuel Category.

      $$\text{Minimum Burnup} = 1{,}000 \times [A_1 \times En^3 + A_2 \times En^2 + A_3 \times En + A_4]$$

   e. All relevant uncertainties are explicitly included in the criticality analysis. No additional allowance for burnup uncertainty or initial enrichment uncertainty is required.

Figure 3.7.17-1 (page 3 of 3)

Notes (continued):

f. Conservatively low values of Decay Time may be used to calculate the Minimum Burnup value, or interpolation may be used. If interpolation is used, Minimum Burnup values for tabulated Decay Time values above and below the actual value should first be determined. Next, linear interpolation between these values may be used to determine the Minimum Burnup value. No extrapolation beyond 20 years is permitted.

g. Initial Enrichment (En) is the nominal U-235 weight percent enrichment of the central zone region of fuel, excluding axial blankets, prior to fuel depletion.

h. If the computed Minimum Burnup value $\leq$ 0 MWD/MTU, the Minimum Burnup Requirement is > 0 MWD/MTU.

5. In all Arrays, an assembly with a higher Fuel Category number can be utilized in place of any fuel assembly with a lower Fuel Category Number, with the following exception.

   a. Fuel Group F1 assemblies are not allowed to be stored in Array II-C, regardless of Fuel Category.

6. An empty (water-filled) cell can be substituted for any fuel-containing cell in all storage arrays.

7. Any storage array location designated for a fuel assembly can be replaced with non-fissile hardware. Items other than Fuel Assemblies which contain fissile material are restricted to storage in Region I.

8. Fuel assembly inserts approved for use in the reactor core can be inserted in a stored assembly (in the Spent Fuel Pool) without affecting the fuel category.

Table 3.7.17-1

Minimum Applicable Initial Enrichment for
Table 3.7.17-2 and Table 3.7.17-3 Fitting Coefficients

Fuel Category	Fuel Group F1	Fuel Group F2
6	1.25	1.20
5	N/A	1.30
4	1.35	1.45
3	N/A	1.45
2	N/A	3.55

Table 3.7.17-2

Fuel Group F1
Nominal Fuel Rod Outer Diameter of 0.374"

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu)
as a Function of Decay Time and Initial Enrichment (En)

Fuel Category	Decay (yrs)	Fitting Coefficients			
		A_1	A_2	A_3	A_4
6	10	1.4351	-17.3247	73.3805	-67.4585
6	15	1.7078	-18.7916	74.6322	-67.2637
6	20	0.5289	-9.9969	53.7741	-52.6302
4	10	-0.0444	-1.3474	22.7039	-28.0852
4	15	0.2015	-2.6257	24.1016	-28.2473
4	20	0.4646	-4.1432	26.3891	-29.2170

Note: Fuel must have at least 10 years of decay time and less than or equal to 3.5 weight percent Initial Enrichment to utilize Table 3.7.17-2.

Table 3.7.17-3

Fuel Group F2
Nominal Fuel Rod Outer Diameter of 0.360"

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu
as a Function of Decay Time and Initial Enrichment (En)

Fuel Category	Decay (yrs)	Fitting Coefficients			
		A_1	A_2	A_3	A_4
6	0	0.5789	-7.4498	42.4056	-41.1591
6	5	0.5247	-6.8992	39.7676	-38.6927
6	10	0.2701	-4.4306	31.9841	-32.4674
6	15	0.3105	-4.5582	31.1825	-31.3916
6	20	0.2374	-3.8754	28.8900	-29.4975
5	0	0.9373	-11.2553	54.7226	-54.1769
5	5	0.6169	-8.1494	44.7801	-45.7968
5	10	0.5380	-7.1852	40.7044	-41.9545
5	15	0.5385	-7.0180	39.2299	-40.3213
5	20	0.5200	-6.7906	38.0244	-39.0979
4	0	0.2553	-3.9826	30.6152	-36.7967
4	5	0.2366	-3.6430	28.2160	-33.9749
4	10	0.4387	-5.6018	33.3609	-37.9327
4	15	0.5450	-6.6302	36.0760	-40.0315
4	20	0.6327	-7.4663	38.2724	-41.7257
3	0	0.5317	-6.1006	32.7118	-36.2263
3	5	0.5228	-5.9434	31.2846	-34.4602
3	10	0.5432	-6.1075	31.1578	-33.9933
3	15	0.5206	-5.8897	30.1768	-32.9600
3	20	0.5158	-5.7796	29.4050	-32.0577
2	0	0.0000	1.6738	-8.5396	9.2206

TS 4.3.1, Criticality

Current TS 4.3.1.1.c states:

> $k_{eff} \leq 0.95$ if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

Revised TS 4.3.1.1.c would state:

> $k_{eff} \leq 0.95$ if fully flooded with water borated to 400 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

Current TS 4.3.1.1.e states:

> A nominal 10.6 inch by nominal 11 inch center to center distance between fuel assemblies placed in Region I fuel storage racks;

Revised TS 4.3.1.1.e would state:

> A nominal 10.65 inch by nominal 11.05 inch center to center distance between fuel assemblies placed in Region I fuel storage racks;

Current TS 4.3.1.1.f states:

> New or partially spent fuel assemblies may be allowed restricted storage in a 1 out of 4 configuration in Region II fuel storage racks (as shown in Figure 3.7.17-4) or unrestricted storage in Region I fuel storage racks;

Revised TS 4.3.1.1.f would state:

> New or partially spent fuel assemblies may be allowed restricted storage in a 1 out of 4 configuration in Region II fuel storage racks (as shown in Figure 3.7.17-1, Array II-E) or unrestricted storage in Region I fuel storage racks;

Current 4.3.1.1.g states:

> New or partially spent fuel assemblies with a discharge burnup in the "acceptable" domain of Figure 3.7.17-1 may be allowed unrestricted storage in a 4 out of 4 configuration in Region II fuel storage racks as shown in Figure 3.7.17-4;

Revised TS 4.3.1.1.g would state:

> Storage of new or spent fuel assemblies in Region II storage racks must comply with 3.7.17 Spent Fuel Assembly Storage.

In addition, current TS 4.3.1.1.h and TS 4.3.1.1.i would be deleted.

New TS 5.5.22, Spent Fuel Storage Rack Neutron Absorber Monitoring Program

New TS 5.5.22 would be added as follows:

Spent Fuel Storage Rack Neutron Absorber Monitoring Program

The Region I storage cells in the CPNPP Spent Fuel Pool utilize the neutron absorbing material BORAL, which is credited in the Safety Analysis to ensure the limitations of Technical Specification 4.3.1.1 are maintained.

In order to ensure the reliability of the Neutron Poison material, a monitoring program is required to routinely confirm that the assumptions utilized in the criticality analysis remain valid and bounding. The Neutron Absorber Monitoring Program is established to monitor the integrity of neutron absorber test coupons periodically as described below.

A test coupon "tree" shall be maintained in each SFP. Each coupon tree originally contained 8 neutron absorber surveillance coupons. Detailed measurements were taken on each of these 16 coupons prior to installation, including weight, length, width, thickness at several measurement locations, and B-10 content (g/cm^2). These coupons shall be maintained in the SFP to ensure they are exposed to the same environmental conditions as the neutron absorbers installed in the Region I storage cells, until they are removed for analysis.

One test coupon from each SFP shall be periodically removed and analyzed for potential degradation, per the following schedule. The schedule is established to ensure adequate coupons are available for the planned life of the storage racks.

Year	Coupon Number	Year	Coupon Number
2013	1	2028	5
2015	2	2033	6
2018	3	2043	7
2023	4	2053	8

Further evaluation of the absorber materials, including an investigation into the degradation and potential impacts on the Criticality Safety Analysis, is required if:

- A decrease of more than 5% in B-10 content from the initial value is observed in any test coupon as determined by neutron attenuation.
- An increase in thickness at any point is greater than 25% of the initial thickness at that point.

3.3 NRC Staff Evaluation

3.3.1 SFP NCS Analysis Review

A main objective of a SFP NCS analysis is to determine limiting fuel mechanical and operational characteristics that will maximize the reactivity of spent fuel stored in the SFP so that the minimum subcritical margin can be determined. This can be achieved with the central idea of maximizing spectral hardening in combination with minimizing U-235 depletion. Specifically, spectral hardening efficiently creates fissile plutonium isotopes, which increases the reactivity of the fuel assembly when stored in the SFP. Several mechanisms cause spectral hardening such as lower water moderator densities, which decreases neutron thermalization, ultimately decreasing thermal U-235 fission and leading to increased plutonium creation. The presence of burnable absorbers, whether they are integral to the fuel or inserted into guide tubes, displaces water, which again reduces neutron moderation, and allows for more fissionable U-238 to be converted to fissile plutonium. By their nature, burnable absorbers are efficient at absorbing thermal neutrons, preventing them from reaching the fuel, and reducing the consumption of fissile isotopes. Since burnable absorbers preferentially absorb thermal neutrons, rather than fast neutrons, this further enables conversion of U-238 to fissile plutonium. Higher fuel temperatures also have a similar effect as this increases Doppler broadening in the U-238 fuel resonance region of the absorption cross-section, increasing the probability of capturing neutrons, and converting U-238 to fissile plutonium. These, and other, considerations, such as the reactivity effects of manufacturing tolerances, are reviewed with increased scrutiny by the NRC staff to ensure that the analysis assumptions are reasonably conservative.

The codes used – both the depletion code and criticality code – should also be verified and validated in order to establish a high degree of confidence in the numerical results. The validation should typically be based on actual experimental results that take into account all relevant characteristics of the application of interest. For the depletion code validation, there is not a generally accepted method for determining code biases and bias uncertainties due to fuel depletion; therefore, engineering judgment is relied upon. For the criticality code validation, there are extensive sets of laboratory critical experiments, but these experiments are not generally comprehensive for most applications. To treat validation gaps, additional uncertainties are typically incorporated into minimum subcritical margin assessments.

3.3.1.1 Selection of Bounding Assembly

CPNPP has operated with multiple fuel designs by multiple vendors, but all contain a 17 x 17 array of fuel rods containing 24 guide tubes and one instrumentation tube; some designs use Zircaloy fuel rod cladding and some use ZIRLO. For the NCS analysis, the various fuel designs are categorized into one of two groups based on two different nominal fuel rod outer diameters. Within the first group (Group F1), Westinghouse Electric Company LLC (Westinghouse) has identified two distinct fuel designs, which have almost identical, neutronically important mechanical properties except for guide and instrument tube diameters. Consequently, a single set of mechanical properties are used when developing isotopics for Group F1 fuel. Within the second group (Group F2), there is more mechanical property variation among one of the three fuel types within the group than the others, [[

]] The primary difference between Group F1 and Group F2 is the fuel pellet diameter.

Due to variations in fuel management over the history of CPNPP operation, in order to minimize the number of depletion calculations required to determine a single bounding fuel design for each fuel group based on mechanical properties, the licensee decided only to include [[

]] The NRC staff notes that [[

]] for a given fuel design is appropriate and conservative. For Group F1, a total of three unique fuel designs were identified, as given in Table 3-6, "Group F1 Fuel Design Parameters," of WCAP-17728-P, and for Group F2, nine were identified, and are given in Table 3-7, "Group F2 Fuel Design Parameters," of WCAP-17728-P (Reference 20).

The Group F1 and F2 fuel designs were further pared down so that a single limiting fuel design for each group was determined. In WCAP-17728-P, Section 4.3.1, "Group F1 Fuel Design Selection," it is stated that "the Group F1 fuel designs presented in Table 3-6 are consolidated into a single fuel design [[

]] Since only one fuel design in Group F1 has axial blankets, it would be conservative to select the limiting profile from the fuel design with blankets as the lower enriched fuel at the top of the assembly would cause a significant depression in the axial burnup profile that would not be as significant in the non-blanketed fuel. For Group F1, [[

]] Additionally, the Pyrex burnable absorber contains 24 fingers versus 12 fingers in the WABA design further increasing the amount of spectral hardening.

For Group F2 fuel, of the originally identified nine fuel designs, only four were considered. The only design that does not include axial blankets was not included. The licensee stated that [[

]] This has been clarified in Attachment 2 of the licensee's December 17, 2013, response to request for additional information (RAI), Question 1 (RAI-1). [[

]] The other two fuel designs considered are distinct from all of the others. In RAI-2 dated November 19, 2013 (Reference 21), the NRC staff requested the licensee to clarify why certain allowable burnable absorber combinations were excluded from the limiting Group F2 fuel assembly evaluation without a formal analysis. In the December 17, 2013, response to RAI-2, the licensee performed several evaluations for additional burnable absorber combinations in question and showed that the assumptions for the design basis Group F2 fuel assembly remained bounding for all past use fuel assemblies and for current and future fuel assemblies, when the WABA length is restricted to 120 inches rather than the 132-inch design that was analyzed in the design basis calculation. This limitation will ensure that the other burnable absorber

combinations remain bounded for all current and future cycles of operation. The licensee has added Commitment Number 4760732 to restrict WABA lengths exceeding 120 inches from being used at CPNPP for all future cycles (Reference 5).

The NRC staff notes that crediting the blankets for Group F2 fuel in the criticality analysis restricts the applicability of the criticality analysis to fuel with axial blankets less than or equal to 2.6 weight percent (wt%) uranium dioxide (UO_2) and axial length greater than or equal to 6 inches for Group F2 when crediting burnup. As discussed in the licensee's December 17, 2013, response to RAI-18, any future fuel design that exceeds this credited blanket enrichment, or is shorter than the analyzed axial length, will require a new depletion and criticality analysis, demonstrating that the fuel assembly outside of the analysis area of applicability (AOA) can be safely stored in either CPNPP SFP in a storage cell within an array that implements burnup credit. As discussed in the response to RAI-18, outlier fuel based on fuel assembly blanket characteristics can be safely stored as fresh fuel (i.e., either in Region I or Array II-E of Region II) as long as the first five parameters in WCAP-17728-P, Table 6-16 are satisfied. Furthermore, as discussed in the response to RAI-4, in the licensee's response to the request for supplemental information dated July 16, 2013 (Reference 2), an outlier fuel assembly based on depletion characteristics will be restricted to either Region I or to Array II-E in Region II if stored in the SFP.

Isotopics were generated in [[

]]

Based on the above, the NRC staff concludes that the licensee's approach for selection of bounding assemblies is acceptable.

3.3.1.2 Depletion Analysis

3.3.1.2.1 Depletion Code Validation

The licensee used the PARAGON code to calculate the isotopic composition of the spent fuel as a function of fuel burn up, initial feed enrichment, and decay time. The NRC staff has approved PARAGON for PWR depletion calculations as a part of its approval of WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," dated March 18, 2004 (Reference 22). The uncertainty in the calculated k_{eff} introduced by the isotopic depletion uncertainty was addressed by applying 5 percent of the reactivity decrement from depletion as an uncertainty component in the determination of the maximum k_{eff}. The NRC staff concludes that this uncertainty treatment is acceptable because it is consistent with ISG DSS-ISG-2010-01.

3.3.1.2.2 Depletion Parameters

In WCAP-17728-P, Section 4.1, "Depletion Modeling Assumptions," the general modeling assumptions are listed and include: [[

]]

The licensee uses the guidance in NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," February 2000 (Reference 23), to appropriately develop the modeling assumptions for several of the design basis reactor operation parameters. Guidance from ISG DSS-ISG-2010-01 and NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," March 2003 (Reference 24), is also appropriately used.

Soluble Boron Modeling

To account for the presence of soluble boron during reactor operation, the licensee uses the guidance in NUREG/CR-6665, which discusses the appropriateness of using a constant cycle average soluble boron concentration rather than the actual boron letdown curve. Since the licensee is using an average boron concentration that conservatively bounds all past and current cycle average soluble boron concentrations for both units, the NRC staff determined that this assumption is conservative.

Fuel and Moderator Temperature

Fuel temperature modeling is important as higher fuel temperatures lead to increased spectral hardening. The reactivity effect increases in importance with increased fuel depletion as discussed in NUREG/CR-6665. In WCAP-17728-P, fuel temperatures are determined [[

]]

[[

]] Maximizing moderator temperature is conservative primarily because the moderator density is decreased, which causes increased spectral hardening. [[

]]

In the April 17, 2014, response to RAI-4, the licensee clarified that the moderator density used in the PARAGON depletion calculations [[

]] the licensee has shown that the moderator density has been treated appropriately (Reference 5 dated December 17, 2013).

Maximum Average Assembly Power and Operating History Effects

All assemblies in the NCS analysis have a high radial peaking factor applied to the nominal assembly power during depletion, in order to conservatively maximize the fuel temperature. As noted in the analysis AOA, past, current, and future fuel designs are limited by the corresponding maximum burnup averaged assembly power. As discussed in WCAP-17728-P, Section 4.2.2.3, "Operating History and Specific Power," ISG DSS-ISG-2010-01 states that when it is not physically possible for a fuel assembly to simultaneously experience two bounding values, the dominate parameter should be maximized and the nominal value should be used for the subordinate parameter. Since it is expected that assuming a conservatively high fuel temperature will result in a more conservative effect than assuming a conservatively low assembly average specific power, the nominal assembly average specific power should be used in the analysis, however this was not done. **[[**

]]

The assumed assembly average specific power and operating history has a minimal effect on the assembly reactivity in the SFP as noted in NUREG/CR-6665, which further indicates that the in-rack k_{eff} could vary by hundreds of percent millirho (pcm) based on sensitivity studies considering a range of operating histories. The licensee has added a **[[**

]] and the NRC staff has reasonable assurance that this is adequate to offset the potentially non-conservative selection of a high specific power when a nominal value should have been used.

Rodded Operation

WCAP-17728-P, Section 5.8, "Rodded Operation," states that rodded operation has not been considered in the depletion analysis since "Comanche Peak has not operated at full power with control rods inserted a significant length...nor are there plans to begin operating in such a manner." Despite there not being a history or plans for rodded operation, rodded operation has been added to the analysis AOA and is not to exceed 0.1 GWd/MTU per cycle. The NRC staff agrees that the analysis AOA limit of rodded insertion up to a burnup of 0.1 GWd/MTU/cycle is bounded by the current analysis, since the short duration would not have an appreciable effect on the discharge reactivity of any fuel assembly in terms of spectral hardening and impact on the axial burnup distribution; consequently, the rodded operation restriction is appropriate for past and current CPNPP cycles.

In the December 17, 2013, response to RAI-19 clarifying the licensee's treatment of rodded operation with respect to the corresponding TS SR 3.7.17.1, the licensee's original understanding was that outlier fuel (i.e., fuel deviating from the AOA) would be treatable under the provisions of 10 CFR 50.59, allowing for the evaluation of outlier fuel for storage in the SFP as discussed in WCAP-17728-P, Section 6.2.1, "Outlier Assemblies," without NRC approval (Reference 5). However, since the AOA is part of the proposed methodology, any deviation from it would be considered a change to the methodology, which would require NRC approval. In the response to RAI-19, the licensee confirmed its position in WCAP-17728-P, Section 6.2.1 that fuel assemblies that do not meet the 0.1 GWd/MTU/cycle limit **[[**

]] This is acceptable since any reactivity increase due to control

rods inserted into a fuel assembly will be less in magnitude than the reactivity decrease associated with crediting burnup without control rods inserted into a fuel assembly. Therefore, not crediting burnup when rods are inserted past 0.1 GWd/MTU in a given cycle is conservative and acceptable when determining the fuel assembly burnup for use with TS SR 3.7.17.1. This treatment of rodded operation was added as licensee Commitment Number 4760741.

In the December 17, 2013, response to RAI-19, the licensee also stated that a detailed review of past operation indicated that several cycles would not satisfy the 0.1 GWd/MTU/cycle limitation in the proposed AOA; therefore, fuel assemblies from these cycles would also be treated in the same manner as future cycles that exceed the 0.1 GWd/MTU/cycle limit. Additionally, the licensee stated that "the administrative controls and Configuration Confirmation Software tools...will incorporate limitations to ensure that the appropriate burnup is credited for fuel assemblies which have experienced HFP [hot full power] Rodded Operation beyond the low threshold required by the AOA."

Based on the above, the NRC staff concludes that the selection of the depletion parameters is acceptable.

3.3.1.2.3 Axial Burnup Profiles

Selection of a set of limiting axial burnup profiles (i.e., in this case, a set based on different burnup categories) for a given fuel design is one of the most important aspects of determining the minimum subcritical margin in the SFP for analyses that use burnup credit. This is due to the axial end effect phenomenon that becomes more prominent with increasing fuel assembly burnup. This phenomenon has been discussed in detail in various technical publications covering burnup credit for LWR fuel including NUREG/CR-6665.

In WCAP-17728-P, the general process starts with a survey of **[[**

]] as discussed above in Section 3.3.1.1 of this safety evaluation.

Based on the above, the NRC staff concludes that the axial burnup profiles selected are acceptable.

3.3.1.3 Criticality Analysis

3.3.1.3.1 Criticality Code Validation

For the criticality calculation, the licensee used SCALE Version 5.1, with the 44-group Evaluated Nuclear Data File, Version 5 neutron cross-section library. SCALE is a comprehensive modeling and simulation suite for nuclear safety analysis and design, developed, and maintained by Oak Ridge National Laboratory (ORNL) under contract with the NRC and the U.S. Department of Energy to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs (Reference 27). This computer code and the corresponding nuclear data set used in this application have been used in many NCS analyses, and is an industry standard. Therefore, the NRC staff considers their use in the current application to be acceptable.

The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainty are determined for use in the criticality calculation. The ISG DSS-ISG-2010-01 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (Reference 28). NUREG/CR-6698 states, in part, that:

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A sufficient number of experiments with varying experimental parameters should be

selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698 as guidance for review of the code validation methodology presented in the application. The basic elements of validation are outlined in NUREG/CR-6698, including identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the AOA.

SCALE which was used in both the code benchmark analysis and the fuel storage analysis, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-III, and KENO V.a (KENO). The licensee performed the validation of the CSAS25 sequence by comparing KENO calculated k_{eff} values with several different sets of critical configurations depending on the type of storage configuration being analyzed; a total of [[ ]] critical configurations were included. [[

]] The sources of critical configurations are the *International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE) (2009 Edition)* (Reference 29), NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," March 1997 (Reference 30), and NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," September 2008 (Reference 31).

The IHECSBE was prepared by a working group comprised of experienced criticality safety personnel from the United States, the United Kingdom, Japan, the Russian Federation, France, Hungary, the Republic of Korea, Slovenia, Serbia, Kazakhstan, Israel, Spain, Brazil, the Czech Republic, Poland, India, Canada, China, Sweden, and Argentina. The handbook contains criticality safety benchmark specifications that have been derived from critical experiments that were performed at various experimental facilities around the world. The benchmark specifications are intended for use by criticality safety engineers to validate calculational techniques used to establish minimum subcritical margins for operations with fissile material. Therefore, the NRC staff considers the IHECSBE to be an appropriate source of information for the critical experiment models. Critical experiments from NUREG/CR-6361 contain important features such as soluble boron and poisoned fuel rods. The use of the HTC experiments documented in NUREG/CR-6979 is important to cover the major actinide distribution of burned fuel. The NRC staff has reviewed the experiments used in the validation of the criticality code for the CPNPP SFPs and considers them appropriate for that use.

WCAP-17728-P states that [[

]] are explicitly represented in the burnup credit analysis. Since only a subset of fission products are credited in the criticality analysis, this is conservative as credit for additional fission products would reduce the reactivity of the system. An additional conservatism, as explained in the licensee's December 17, 2013, response to RAI-3, is the [[

]]

To compensate for a lack of critical experiments containing fission products, a fission product worth uncertainty is added as [[

]] 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 (Reference 32), 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 licensee's approach would produce a non-conservative uncertainty estimate by approximately 100 to 200 pcm, but it could be larger. Since the approach for determining the isotopic validation gap uncertainty only covers fission products and does not include all minor actinides and since the approach is based on preliminary research, in RAI-5 dated November 19, 2013 (Reference 21), the NRC staff asked the licensee to provide justification for not including the minor actinides and for treating the resultant term as an uncertainty rather than a bias.

In its December 17, 2013, response to RAI-5, the licensee stated that several conservatisms can be credited in order to conform to the recommended treatment of the minor actinide and fission product validation gap (Reference 5). [[

]] For each case, there was a net reactivity increase. Part of the licensee's approach to compensate for each reactivity increase was to [[

]] Section 4.2.2, "Moderator Temperature/Density," from NUREG/CR-6665 provides reactivity effects due to temperature that support the reactivity effects calculated by the licensee (Reference 23). As part of Commitment 4760732, the maximum inlet temperature is limited to less than or equal to 561.7 °F for all future fuel cycles, and is consistent with the credit taken in the above analysis.

For the three remaining cases where crediting moderator temperature conservatisms could not fully compensate for the reactivity increase due to applying the recommended validation gap bias, other conservatisms were credited. For one of the cases [[

]]

For two of the three remaining cases mentioned above, additional credit is taken for conservative treatment of the theoretical fuel density (Group F2 fuel only) in WCAP-17728-P. In its December 17, 2013, response to RAI-5, the licensee stated that the depletion analyses in WCAP-17728-P use a fuel theoretical density that is [[ ]] and reducing the fuel theoretical density of these two cases provides the remaining margin needed to fully compensate for not using the recommended 1.5 percent bias discussed above. Since Commitment 4760732 also limits the 95/95 upper bound fuel theoretical density to be less than 96.5 percent for all current and future fuel assemblies, credit for a fuel theoretical density that is [[ ]] less than that assumed in the WCAP-17728-P analysis is acceptable for Group F2 fuel. Based on the revised treatment of the moderator temperature, [[ ]] and Group F2 fuel theoretical density, in addition to Commitment 4760732 as discussed above, the licensee has demonstrated that there is enough margin in the WCAP-17728-P analysis to fully compensate for not applying the 1.5 percent recommended bias to account for the minor actinide and fission product validation gap.

The licensee identified the applicable operating conditions for the validation (e.g., fissile isotope, enrichment of fissile isotope, fuel chemical form, types of neutron absorbers, moderators and reflectors, range of moderator to fissile isotope, and physical configurations) consistent with NUREG/CR-6698. The methodology that the licensee used to determine the bias and bias uncertainty to apply to the results of the KENO calculations essentially followed NUREG/CR-6698; however, there were some deviations – mainly with the method used to calculated bias and bias uncertainty for cases where statistically significant trends are found. In WCAP-17728-P, Section A.5.2, "Determination of Bias and Bias Uncertainty and Normality Check," the licensee performs data normality testing and calculates [[

]] The uncredited administrative margin also provides reasonable assurance that subcritical margin exists and that the regulatory k_{eff} limit is not exceeded.

[[

]]

Based on the above, the NRC staff concludes that the information supporting the criticality code validation for the CPNPP SFPs is acceptable.

3.3.1.3.2 Normal Conditions

CPNPP has separate SFPs for Unit 1 and Unit 2 that both employ two distinct modular storage rack designs throughout the pool. Region I of the CPNPP SFPs was designed to hold fresh fuel and utilizes a flux trap design which uses two Boral panels between each assembly to reduce the neutronic interaction between adjacent assemblies. The other modular storage rack design is used in Region II, which is a high density rack design meant primarily for the storage of spent fuel, and is used in Region II of both CPNPP SFPs. The intent of the Region II design was to utilize a single Boraflex panel between each assembly; however, due to issues identified with Boraflex degradation before installation of the inserts, the Boraflex inserts were never installed, and therefore they are not credited in the analysis. In Region II of the Unit 1 SFP, the panels were removed from the wrappers before rack installation, but for Region II of the Unit 2 SFP, the wrappers were never installed. The licensee notes that the Region II analyses (specifically for Unit 1) conservatively do not model the Boraflex wrappers since "the wrappers displace moderator and are made of stainless steel which absorbs neutrons." The NRC staff determined that assumption to be appropriate and conservative since modeling the stainless steel as water would cause a net increase in k_{eff}; primarily since stainless steel is a stronger absorber than water in this application.

The criticality analysis is based on an infinite array of storage cells, which [[

]] Credit is taken for the SFP wall in the analysis of Array II-A, which is discussed further in Section 3.3.1.3.3 below. Since credit is taken for the wall for Array II-A, the NRC staff had additional questions regarding the treatment of the Array II-A boundary conditions. [[

]] however, this would not be appropriate for Array II-A due to the asymmetrical geometry in the x-direction caused by the SFP wall and Array II-B interface requirement. Consequently, the NRC staff issued RAI-1 dated February 27, 2014, for clarification of the Array II-A boundary conditions [[

]] In its April 17, 2014, response to this RAI, the licensee discusses the boundary conditions and model characteristics used in detail for Array II-A in the KENO model (Reference 7). [[

]]

RAI-2 dated February 27, 2014, covers similar modeling issues regarding the Region I-to-Region II interface. In reviewing the licensee's April 17, 2014, response to RAI-2, the NRC staff also determined that the Region I-to-Region II interface modeling is acceptable since an

appropriate number of fuel assemblies were included in the KENO model, in both the x- and y-directions, in conjunction with appropriate model boundary conditions.

During normal operation, at least 2400 parts per million (ppm) of soluble boron is present in the SFPs, as reflected in the proposed revised TS 3.7.16, and the moderator temperature is less than or equal to 150 °F. The limiting normal operation soluble boron concentration requirement is 400 ppm, and does not credit any of the remaining 2000 ppm of soluble boron that would be present. The licensee models the isotopic fraction of B-10, the principal neutron absorber, as 0.194. A technical report published in 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 (Reference 35). Since the licensee has determined that 400 ppm of soluble boron is required to meet the applicable regulatory limit even though 2400 ppm is present, the normal condition analysis shows that considerable margin is available to negate any non-conservative reactivity effects due to the assumption of a B-10 isotopic fraction of 0.194, which could possibly be lower. However, 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, the NRC staff issued RAI-7 asking the licensee to justify the B-10 isotopic fraction of 0.194.

In the December 17, 2013, response to RAI-7, the licensee reviewed the past isotopic analysis reports of B-10 composition, required from the manufacturer, and showed that the minimum B-10 atom fraction over the 20-year period between 1992 and 2012 was 0.1973, demonstrating that the WCAP-17728-P assumption of 0.194 is conservative (Reference 5). Additionally, the current B-10 atom fraction in both CPNPP SFPs was verified to be above the WCAP-17728-P assumptions with values of 0.1982 and 0.1977. The licensee also discussed the potential for decreases in B-10 atom fraction due to various sources. The sources identified are mixing of depleted boron from the RCS and the SFP water, addition of fresh boric acid to the SFP, and depletion of B-10 due to neutron interactions in the SFP. The licensee explained that it is not possible to achieve a B-10 atom fraction below the 0.194 value assumed in the criticality safety analyses in detail in the response to RAI-15, which basically uses B-10 depletion data from [[

]] The calculation shows that this conservative and unlikely scenario would result in a minimum B-10 atom fraction of 0.195 and would still maintain margin to the 0.194 value assumed in the criticality safety analyses. The licensee has also added a Commitment Number 4760737 to review the B-10 concentration in the RCS during each refueling outage after borating to the SFP minimum soluble boron concentration, and if the calculated B-10 atom fraction is below 0.194, a B-10 isotopic measurement will be performed on the SFP after adequate mixing occurs, but prior to the next refueling outage, to ensure that the B-10 atom fraction in the SFP is not reduced below 0.194. Similar to the measurement commitment made regarding RCS water mixing with the SFP water during an outage, Commitment Number 4760738 was also made to ensure that any abnormal boration of the SFP is monitored to ensure that the B-10 atom fraction is not lowered below 0.194. The commitment states that if a review of the "SFP Boron Measurement" history, performed each refueling outage, shows that the SFP boron

concentration increases by more than 100 ppm, a review of B-10 atom fractions for boric acid purchased at CPNPP will be performed. If the review shows that the boric acid has a B-10 atom fraction below 0.194, a B-10 measurement will be performed on the SFP prior to the next refueling outage to ensure the assumptions of the criticality safety analysis remain valid. Based on the plant historical records, the analyses performed, and the commitments made, the NRC staff determined that the assumption of a B-10 isotopic fraction equal to 0.194 in the CPNPP criticality safety analysis is conservative and appropriate.

In WCAP-17728-P, Section 5.4, "Interface Conditions," the methodology for addressing the interface between rack designs, between storage arrays within a rack design, and within a storage array itself are not consistent with the guidance in ISG DSS-ISG-2010-01, which states that "absent a determination of a set of biases and uncertainties specifically for the combined interface model, use of the maximum biases and uncertainties from the individual storage configurations should be acceptable in determining whether the k_{eff} of the combined interface model meets the regulatory requirements." Consequently, the NRC staff issued RAI-3 dated February 27, 2014, requesting that the licensee either revise the analysis to be consistent with ISG DSS-ISG-2010-01 or provide the justification for the methodology used in WCAP-17728-P including the results of the analyses performed to support that methodology.

In its April 17, 2014, response to RAI-3, the licensee provided the results of analyses performed to support an alternative methodology than that given in ISG DSS-ISG-2010-01. The licensee analyzed a representative subset of **[[**

]] Based on the discussion above regarding the licensee's April 17, 2014, response to RAI-2, the NRC staff determined that the modeling of intra-array, intra-region, and inter-region interfaces is acceptable.

In WCAP-17728-P, Section 5.5.2, "Type 2 Normal Conditions," the licensee describes evolutions where fuel is removed from the storage rack for a specific procedure, such as cleaning, inspection, reconstitution, and sipping; and then reinserted into the storage rack. For criticality control, in all instances "only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other

assemblies [for inspection cells bordering Region II]," therefore, these scenarios are bounded by the normal condition criticality analysis for inspection cells in Region II. There is no assembly pitch separation requirement for Region I since any single fuel assembly located in an inspection cell bordering Region I is already bounded by the Region I criticality safety analysis as noted in the December 17, 2013, response to RAI-8.

Based on the above, the NRC staff concludes that the licensee's evaluation of the normal conditions in the CPNPP SFPs is acceptable.

3.3.1.3.3 Abnormal Conditions

Section 5.7 of WCAP-17728-P presents the abnormal conditions considered in the analysis. The licensee considered the following abnormal conditions: (1) misloaded assembly, (2) SFP temperature greater than operating range, (3) dropped and misplaced fresh fuel assembly, and (4) seismic accident.

The boron dilution analysis conclusions have not changed with this LAR, therefore, the analysis was not revised. The dilution volume required to dilute the proposed new TS 3.7.16 soluble boron concentration to the new limiting concentration analyzed in the updated criticality safety analysis has increased relative to that previously approved, thereby increasing the time available for intervention.

The licensee modeled a multiple fuel assembly misload event where several assemblies are misloaded in series due to a common cause. This was modeled by [[

]] The licensee's criticality safety analysis determined that the limiting [[ ]] would require full credit of the 2400 ppm of soluble boron required by the proposed new TS limit to comply with the regulatory k_{eff} limit of 0.95. The licensee also performed a single assembly misload analysis, which shows that 1867 ppm of soluble boron is required for the configuration identified as limiting in the multiple misload analysis. Additionally, the single misload analysis k_{eff} was approximately 400 pcm less than the multiple misload analysis, demonstrating that the amount of soluble boron required would actually be less than 1867 ppm under the single misload scenario.

Events where multiple fuel assemblies have been misloaded in the SFP have occurred in the industry. To address this issue, the licensee has proposed additional training and administrative controls (discussed in further detail in Sections 3.3.5 and 3.3.6 of this safety evaluation) to reduce the likelihood of a multiple fuel assembly misload event from occurring. The NRC staff agrees that the additional administrative controls will reduce the likelihood of a multiple fuel assembly misload event from occurring. Furthermore, the licensee's analysis includes a scenario with multiple misloaded fuel assemblies where the licensee's analysis found that it would meet the requirements of 10 CFR 50.68 while crediting the TS required 2400 ppm of soluble boron. The NRC staff has performed confirmatory analyses for other multiple fuel assembly misloading scenarios and believes that, with the additional administrative controls, the

likelihood of a limiting multiple fuel assembly misload event is sufficiently unlikely that there is reasonable assurance that the requirements of 10 CFR 50.68 are met.

For Array II-A, depicted in WCAP-17728-P, Section 5.2, "Array Descriptions," [[

]] 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 possible, by letter dated November 19, 2013 (Reference 21), the NRC staff asked the licensee to analyze it in RAI-9. In its December 17, 2013, response to RAI-9, the licensee showed that [[ ]] remains limiting. Based on the RAI-9 response, the TS limit of 2400 ppm remains appropriate.

To cover the SFP heat-up conditions, the licensee analyzed the pool with a moderator density of [[

]] The k_{eff} results showed that both cases were bounded by the misload event.

The licensee stated that the dropped fuel assembly is non-limiting [[

]]

WCAP-17728-P, Section 5.7.4, "Seismic Event," discusses the impact on the criticality analysis due to a seismic event. [[ ]] The normal operation analysis results, based on 320 ppm of soluble boron, given in Table 5-20, shows that there is approximately [[ ]] reserve margin to the 0.95 k_{eff} limit counteracting [[

]] Furthermore, this accident is clearly bounded by the misload accidents, which require 2400 ppm of soluble boron. 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. Consequently, in RAI-12 dated November 19, 2013, the NRC staff asked the licensee to explain why full credit of the Region I rack module flux trap gap is appropriate during a seismic event. In the December 17, 2013, response to RAI-12, the licensee explained that the NRC has previously reviewed the structural considerations of the CPNPP SFP storage racks with respect to the effect on the criticality safety analysis as indicated by the NRC staff's safety evaluation related to Amendment No. 87 approving changes to the SFP storage capacity on October 2, 2001 (Reference 36). Therefore, since the March 28, 2013, LAR does not introduce any physical changes to the CPNPP SFP storage racks, the NRC staff's conclusions remain unchanged with respect to the ability of the Region I flux trap gaps to be maintained at the nominal value during a design basis seismic event.

Although the licensee shows close to zero margin available between the amount of soluble boron in the limiting abnormal condition analysis and the amount required by the TSs, the accidents considered are based on the guidance in the Kopp Letter and provide reasonable assurance of safety by demonstrating that k_{eff} will remain less than 0.95 for those accident scenarios considered.

Based on the discussion above, the NRC staff concludes that the licensee's evaluation of the accident conditions in the CPNPP SFPs is acceptable.

3.3.1.3.4 Bias and Uncertainty Analysis

The manufacturing tolerances of the storage racks and fuel assemblies contribute to the system reactivity. Consistent with the Kopp Letter, the determination of the maximum k_{eff} should consider either: (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant tolerance variations in the material and mechanical specifications of the racks.

The licensee's analysis evaluated the following manufacturing tolerance uncertainties: **[[**

]]

To determine the reactivity uncertainty associated with a specific manufacturing tolerance, the licensee used KENO to calculate the k_{eff} for the reference condition and the k_{eff} for the perturbed case. The reference condition is the condition with nominal dimensions and properties with the exception of the **[[** **]]** As explained in the licensee's December 17, 2013, response to RAI-13, a tolerance on the Boral thickness was not specified by the manufacturer, and is therefore modeled with its nominal thickness in all Region I analyses. Consequently, the Region I criticality analyses conservatively assume the **[[**

]] The NRC staff determined that using the **[[**

]] is acceptable in lieu of performing sensitivity evaluations on the B-10 areal density and Boral absorber thickness. All tolerance perturbations were applied in the direction that increases reactivity relative to the nominal condition and were performed separately for each allowable storage array configuration. Based on the considerations discussed above, the NRC staff determined that the licensee's treatment of manufacturing tolerances is acceptable because it is consistent with the guidance in the Kopp Letter.

In WCAP-17728-P, Section 5.3.2.2, "Storage Array Biases & Uncertainties Results," it is explained that the biases and uncertainties are [[

]] The NRC staff determined that this approach is acceptable since all biases and uncertainties are treated appropriately on a case-specific basis and since uncertainties deemed sensitive to changes in the neutron spectrum that occur with burnup are treated in a conservative manner.

In WCAP-17728-P, Section 5.3.2.1.2, "Burnup Measurement Uncertainty," the licensee stated that the burnup measurement uncertainty [[

]] The December 17, 2013, response to RAI-14 (Reference 5), describes the basis for the burnup measurement uncertainty citing several references that support the validity of the values assumed in WCAP-17728-P, which include
[[

]] the NRC staff concludes that the burnup measurement uncertainty treatment is acceptable.

According to the Kopp Letter, the criticality analysis should account for the temperature corresponding to the highest reactivity. The licensee performed the base criticality analyses
[[

]] The NRC staff concludes that this is consistent with the Kopp Letter, and therefore acceptable.

Based on the above, the NRC staff concludes that the licensee's evaluation of the bias and uncertainty analysis for the CPNPP SFPs is acceptable.

3.3.1.3.5 Storage Rack Analysis Considerations

In WCAP-17728-P, Section 5.1.2, "The Impact of Structural Materials on Reactivity," the licensee discussed modeling considerations for structural materials used in the SFPCA. The licensee explained that all structural material, both Zircaloy-4 and stainless steel, is conservatively modeled as [[

]] Since this is not a requirement, but an additional conservatism, the NRC staff determined that this modeling practice is acceptable for this application.

The top and bottom nozzles are [[

]] This is a significant conservatism as it [[

]] therefore, the NRC staff concludes that this assumption is acceptable.

The fuel rod spacer grids and grid sleeves are [[

]]

Accounting for Boral neutron absorber material changes in the Region I SFP storage racks is another phenomenon that must be addressed. Recent operating experience has identified that material geometry changes, in the form of Boral blistering, have occurred in the SFP environment. To account for the possible reactivity effects due to Boral blistering, the licensee performed a series of sensitivity calculations. Two methods were used in the study to quantify the reactivity effects due to Boral blistering.

The first method is [[

]] the results of this method are also used to define the criteria used to show operability in the Boral monitoring and surveillance program discussed in Section 3.3.4 of this safety evaluation. For the first method, [[

]]

The second method is [[

]]

Ultimately, the conclusion made by the licensee is that the Region I storage racks can perform their design function even with [[

]]

In order to ensure that a Boral [[

]] the licensee is implementing a surveillance program in TS Section 5.5, "Programs and Manuals," under Subsection 5.5.22, "Spent Fuel Storage Rack Neutron Absorber Monitoring Program," that will monitor any Boral blisters that form on Boral coupons placed directly in the CPNPP SFPs (Reference 1). The neutron absorber surveillance program is discussed further in Section 3.3.4 of this safety evaluation.

Based on the above, the NRC staff concludes that the licensee's evaluation of the storage rack analysis considerations for the CPNPP SFPs is acceptable.

3.3.2 Burnup Limits for Storage Arrays

In WCAP-17728-P, Section 5.2, "Array Descriptions," the licensee proposes six normal storage configurations for use throughout the CPNPP SFPs defined by combinations of one or more fuel categories. Lower number fuel categories correspond to less reactive configurations and higher number fuel categories correspond to more reactive configurations; this is seen with the increasing minimum burnup requirements for higher category storage configurations, which are provided in WCAP-17728-P, Section 6.1, "Burnup Limits for Storage Arrays." Within each storage configuration, both Group F1 and F2 fuel can be stored, but with differing burnup requirements. The licensee states that all fuel categories will not be utilized by CPNPP. Those not being utilized are for Group F1, Fuel Categories 3 and 5. Since no burnup is required for Group F1, Category 1 and 2 fuel, and Group F2, Category 1 fuel, no burnup limits are presented for these storage array configurations. The burnup and enrichment loading curves (BULCs) used by CPNPP are provided, with example requirements at the analyzed initial fuel enrichments and as a function of decay time, in WCAP-17728-P, Section 6.1, and are the same BULCs implemented in the proposed revised TS 3.7.17. The notes in proposed revised TS 3.7.17 that provide specific guidelines for SFP fuel assembly storage were reviewed by the NRC staff and determined to be appropriate and consistent with the NCS analysis reviewed in WCAP-17728-P.

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 BULCs). In WCAP-17728-P, Section 6.1.3, "Decay Time Interpolation," the licensee explained that linear interpolation between adjacent decay time points for a given fuel category and group is

acceptable since isotopic decay is an exponential function which means that assembly reactivity will decay faster than calculations using linear interpolation would predict. While the change in net reactivity as a function of decay time is an exponential relationship over longer periods of time, for the range of analyzed decay times (i.e., less than 20 years), the overall assembly reactivity is decreasing at close to a linear rate between the adjacent decay time points, therefore linear interpolation is acceptable. The NRC staff issued RAI-17 dated November 19, 2013 (Reference 21), in order to better understand the basis for creating the minimum burnup loading curves, which are a function of both enrichment and decay time. The NRC staff was specifically concerned with the curve fitting process and if the specific enrichment points based on the criticality safety calculations actually pass through the curve or are bounded in some way. In the December 17, 2013, response to RAI-17, the licensee explained that the BULCs are all generated to ensure that the loading curves either pass through the explicit burnup/enrichment points or exceed them (Reference 5). The behavior of the curve between the explicit BULC points is also verified to ensure that the curve does not exhibit decreasing burnup requirements as a function of enrichment. Since the BULC generation method is demonstrated to be conservative with respect to all calculated burnup and enrichment limits, the NRC staff concludes that the method is acceptable.

3.3.3 Analysis Area of Applicability

The analysis AOA is summarized in WCAP-17728-P, Section 6.2, "Analysis Area of Applicability" in Table 6-16, "Analysis Area of Applicability." The table is a result of all criticality safety evaluations performed in the various sections of WCAP-17728-P. The licensee explained that any assembly that falls outside of the limits of the AOA is classified as an outlier assembly and would require special consideration for storage with burnup credit. In the December 17, 2013, response to RAI-19 (Reference 5), the licensee stated that, "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." The only AOA outliers that are excepted from requiring NRC approval are those discussed previously in Sections 3.3.1.1 and 3.3.1.2.2 above.

3.3.4 NRC Staff Evaluation of Neutron Absorber Monitoring Program

3.3.4.1 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

Boral is composed of a core cermet of 50 percent B_4C and 50 percent aluminum and an aluminum alloy Al 1100 outer cladding. B_4C is the constituent in the Boral known to perform effectively as a neutron absorber and Al 1100 cladding is an alloy known for its resistance to corrosion.

The licensee's proposed Neutron Absorber Monitoring Program consists primarily of monitoring the physical properties of the absorber material by performing periodic dimensional and visual checks and monitoring the Boron-10 (B-10) content (g/cm^2) to confirm that the neutron absorption capabilities of the Boral material are being maintained.

3.3.4.1.1 Program Description

The purpose of the licensee's proposed Neutron Absorber Monitoring Program is to characterize certain properties of the Boral to assess the capability of the Boral panels in the racks to continue to perform their intended function. The program will monitor how the Boral absorber material properties change over time under the radiation, chemical, and thermal environment found in the SFP.

Boral coupons fabricated from the material incorporated into the racks are installed on a coupon tree that holds 8 coupons each. The rack cells and the BORAL coupons are exposed to the SFP environment. The coupon trees are and will continue to be placed in each SFP at a location that will ensure that they are exposed to the same environmental conditions as the Boral panels. Coupons will be examined at a frequency outlined in the following table.

Year	Coupon Number	Year	Coupon Number
2013	1	2028	5
2015	2	2033	6
2018	3	2043	7
2023	4	2053	8

The NRC staff has reviewed the proposed program description information. The staff agrees that the Boral coupons are representative and would experience the same environmental conditions as the Boral panels in the SFP. The use of the same Boral material for the coupons that was used to fabricate the panels for the racks ensures that the Boral coupon material is representative of the Boral material in the racks. The staff determined that the coupons experience the same environment as the racks and therefore provide an adequate representation of the BORAL in the racks. Therefore, the staff has reasonable assurance that the Boral coupons are representative of the Boral panels and are experiencing representative environmental conditions in the SFP.

3.3.4.1.2 Monitoring Changes in the Physical Properties and Testing of Coupons

The proposed Neutron Absorber Monitoring Program has one coupon removed from each SFP for testing at a frequency outlined in the above table. The following measurements will be performed of these coupons:

1. Dimensional measurements:

    a. Weight
    b. Length
    c. Width
    d. Thickness

2. Neutron attenuation testing

The licensee's criteria for non-acceptance are as follows:

- A decrease of more than 5 percent in B-10 content from the initial value observed in any test coupon as determined by neutron attenuation.
- An increase in thickness at any point is greater than 25 percent of the initial thickness at that point.

The NRC staff determined that the testing and the criteria for non-acceptance can identify material property changes in the Boral before significant degradation that will affect the criticality analysis since the values that were analyzed in the criticality analysis bounds the criteria for non-acceptance. Also, since the criteria for non-acceptance correlates to the inputs to the criticality analysis, the staff finds the criteria for non-acceptance to be acceptable. Therefore, the staff has reasonable assurance that the criteria for non-acceptance and testing can identify changes in the material property of Boral before significant degradation can impact the criticality analysis.

In addition, the licensee will take actions as outlined in its procedure if the criteria for non-acceptance are met. This includes "if blisters (or other visible signs of degradation) are observed on these samples, the degradation area will be observed and photographed each time the coupon tree is lifted from the storage racks (for coupon removal or replacement) and compared to previous observations to monitor for adverse trends." Since the licensee will be trending blisters and other visible signs of degradation, the staff has reasonable assurance that the degradation and deformation of Boral will be monitored and able to be mitigated to ensure subcriticality requirements.

The licensee also indicated that after testing is completed, the coupons will be returned to the SFP. The licensee stated that "future coupons removed for testing are reinserted into the SFP after testing is complete." The duration of the coupons' removal for testing may be 2-3 months which "is very small in comparison to the lifetime of the Spent Fuel Pool, and therefore the time the coupons spend removed from the pool environment does not have a significant impact on the ability for the coupons to detect signs of degradation." The coupons' "location and move times are tracked for the samples (to enable demonstration of the total duration that test coupons were removed from the pool environment.)" Also, that there will be a "requirement to perform continual testing every 10 years as long as the Region I storage racks are licensed to store fuel." Based on this information, the NRC staff has reasonable assurance that Boral degradation will be able to be monitored for the life of the SFPs.

3.3.4.2 Technical Specifications

Proposed TS 5.5.22, "Spent Fuel Storage Rack Neutron Absorber Monitoring Program" contains information about the Monitoring Program, frequency of inspecting a coupon, and criteria for non-acceptance. This information has been reviewed by the NRC staff for technical merit as documented in Section 3.3.4.1 of this safety evaluation.

3.3.5 Qualified Core Performance Engineers, Fuel Handling Personnel, and Licensed Operator Training

In its application dated March 28, 2013, the licensee stated that the limitations associated with the proposed TS 3.7.17 are more complex than what is currently in existence. Currently, the loading pattern based on the current TS limits can be performed visually by reviewing a color-coded SFP map which provides a straightforward process for recognizing deviations from the storage pattern. The licensee further stated that while the color-coded SFP maps will still be useful, the increased complexity of the proposed TS 3.7.17 limits results in increased difficulty for identifying non-compliant configurations by simple visual methods. The licensee has proposed improvements in the administrative controls and additional training to compensate for these complex spent fuel storage configurations and to minimize the potential for a multiple fuel assembly misload event.

The licensee stated in its January 16, 2014, response to an RAI (Reference 6) that training for qualified Core Performance Engineers (CPEs) will be completed prior to implementation of the proposed TS changes. The licensee also stated that the CPE qualification process at CPNPP will utilize a Subject Matter Expert (SME) review to ensure that personnel are knowledgeable on activities including fuel move planning and the performance of SR 3.7.17.1. The SME Objectives for the Core Performance Monitoring qualification and the Core Performance Fuel Handling qualification have been updated to reflect the proposed changes to the TS, and the process for performing the Surveillance. These changes were completed by the licensee to ensure that all future personnel qualified under this program will demonstrate adequate understanding of the SFPCA and associated limits prior to independently performing or reviewing fuel move plans. Additionally, the licensee completed a Training Needs Analysis as part of the transition to determine the training needs for currently qualified CPEs. The licensee determined in this needs analysis that currently qualified personnel should be trained on the proposed changes to the TS, the supporting criticality analysis, and the implementing procedures and associated Quality Assurance (QA) software. This training is currently being developed, and will be completed for all personnel qualified to prepare or review fuel move plans prior to implementation of the proposed TS changes.

The licensee also determined that additional training was necessary to focus on the changes to the procedure revisions as described in Enclosure 1 to the licensee's letter dated March 28, 2013, and the increased responsibility of fuel handling personnel. CPEs, Licensed Operators, and Fuel Handling qualified personnel will attend this training.

The licensee further stated that the Operating Experience program at CPNPP, regularly reviews industry operating experience, and applicable lessons learned have been incorporated in to site processes and procedures. Since the qualification processes for fuel handling and fuel move planning activities reference current processes and procedures, the lessons learned from the Operating Experience related to the misload accidents are inherently included as part of the qualification process. Therefore, the licensee complies with the requirements of 10 CFR 50.120 and is generally consistent with the guidance provided by SRP Chapter 18, NUREG-0711, and NUREG-1764.

Based on the new administrative controls developed by the licensee and the procedures and training to be implemented to minimize the potential for multiple fuel assembly misloading

events, the NRC staff concludes that the proposed request provides reasonable assurance that the TS changes are acceptable from the human factors point of view.

3.3.6 New Software Program (PETRIFIED) and Improved Administrative Controls to Account for Increased Complexity

In Enclosure 1 to LAR 13-01, the licensee discussed the implementation of a new software program and improved administrative controls that are proposed to ensure that the increased complexity of the Region II storage configurations does not result in an increased risk of TS 3.7.17 non-compliance due to an error made during fuel movement planning.

The current TS 3.7.17 surveillance activities at CPNPP are performed using the QA software program High Density Storage Limits (HDSL) which calculates the allowable storage configurations of the current CPNPP TS 3.7.17 limits. The licensee noted that since this software requires the user to manually create input files and manually review the fuel move plans to ensure that fuel movement and final configurations are compliant with TS requirements, there is a risk of a human performance error which could result in fuel movement that is not compliant with TS 3.7.17 limits. Additionally, the HDSL software does not evaluate spent fuel pool configurations or fuel move sequences.

As part of the transition to the proposed revised TS 3.7.17 limits, the licensee is proposing to replace the HDSL software with the new PETRIFIED (Pool Evaluator for TechSpec Regions Integrated with Fuel Inventory Electronic Databases) software, Revision 0. The licensee stated that PETRIFIED is an improvement over HDSL in that it provides automated verification of fuel movement plans. The users of PETRIFIED will consist solely of the CPE group, since this group is solely responsible for performance of SR 3.7.17.1.

This new configuration confirmation software will obtain input data directly from two other QA software programs currently used at CPNPP: ShuffleWorks and TARPIT (Thermal Assembly Repository Pad Inventory Tracker). ShuffleWorks is a commercial software used at CPNPP for fuel movement planning and SFP configuration control; but does not perform verification of TS 3.7.17 limits. TARPIT is a software program developed internally at CPNPP that is primarily used to verify the acceptability of Dry Cask Canister loading patterns. TARPIT contains a QA-controlled database with fuel information needed to support validation of the dry cask storage limitations. This database will be used by PETRIFIED to determine the Fuel Category for each fuel assembly in the ShuffleWorks database.

The licensee stated that the PETRIFIED software obtains fuel assembly information automatically and directly from the data source, without the need for the user to perform data manipulation or formatting. Additionally, all TARPIT and ShuffleWorks data files, when accessed by the PETRIFIED software, are opened in a "Read Only" capacity, which prevents unintended changes to the support files.

The primary functions which are required to be performed by PETRIFIED are described in the "Software Features" section of Enclosure 1 of the March 28, 2013, LAR. These include:

1. Categorization of each assembly by Fuel Category

2. Determination of acceptability of a current or planned SFP Configuration

3. Validation of ShuffleWorks Sequence Files on a move-by-move basis (to ensure that potential violations are identified)

PETRIFIED will also provide 'FAILURE' notifications in situations which include:

1. A fuel assembly being stored in a configuration not allowed by TS 3.7.17

2. Any necessary Fuel Information missing from the database

3. Any input parameter being outside of the analyzed range

4. Multiple Items stored in the same location in a Configuration File, or a Sequence file which temporarily places more than one item into a location

5. Items in invalid locations for fuel storage

The licensee stated that the PETRIFIED software was developed and tested per the quality standards of the CPNPP Nuclear Software QA Program, and that no software, other than the programming language, was utilized to develop PETRIFIED.

The PETRIFIED software will be installed and executed from a CPNPP shared network drive with restricted access for which only a small group of individuals, including all members of CPE, have read/write access. The licensee stated that a separate PETRIFIED test location is available to test future software revisions, in order to ensure that temporary testing files do not impact the production location. Per the CPNPP Core Performance software control program, when a software revision occurs, any prior revisions of the software must be removed prior to installing the revised version on the shared network drive.

The licensee stated that the PETRIFIED software and associated data files will be controlled per the CPNPP software QA program, which addresses design documentation, cyber security, configuration control, and media/access controls. The CPNPP software QA program requires:

1. Software features independently tested to ensure accuracy and completeness, reliability, functionality, and ease of use prior to approval.

2. Station procedures control all input data files, and require independent review for any changes, including routine updates.

3. Independent testing of any changes to the software prior to approval of the software revision.

The licensee stated that testing was performed per the Core Performance software control procedure to ensure that the software requirements were met and that all other functionality works as designed. The licensee also stated that the individual responsible for developing and implementing the test plan is a fully qualified member of the CPE group, and is qualified to perform SR 3.7.17.1 surveillance and fuel move planning activities.

The licensee performed independent manual verifications and used spreadsheet calculations to verify the outputs of the PETRIFIED software. These included verification that values from the proposed TS Limits were correctly programmed into the software, and that failure "Rules" which were programmed into PETRIFIED were triggered. Additionally, the licensee stated that testing included verification of proper classification of the entire contents of the CPNPP spent fuel, including the contents of both Region I and Region II of both SFPs 1 and 2 (over 2,300 fuel assemblies).

The licensee stated that although extensive testing was performed to demonstrate the accuracy and reliability of PETRIFIED, it is possible that an undetected software error could result in a violation of the proposed TS 3.7.17 limitations, which would be considered a misload accident. The procedural administrative controls described in Enclosure 1 of the March 28, 2013, LAR are meant to limit the type of software misload errors to those bounded by the analyzed cases in Section 5.7 of WCAP-17728-P Revision 1. The licensee notes that if following the steps of the fuel movement plan results in a violation of these requirements, Fuel Handling procedures require that fuel movement must stop, regardless of the instructions provided by the CPE or the SR 3.7.17.1 surveillance results.

Per the CPNPP Core Performance software control procedure, if an error in the software is discovered, the software users will be notified to prevent invalid outputs from being used, and the error will be entered into the CPNPP corrective action program to evaluate Operability and determine the proper corrective actions. Regarding software changes needed to ensure accuracy of surveillance reports, the licensee stated that the changes will be tested per the software QA program and implemented prior to performing further SR 3.7.17.1 surveillances. If it is determined that the error has resulted in a storage configuration, which is non-compliant with LCO 3.7.17, the actions required by the TS will be followed.

The licensee noted that the procedure for performing proposed SR 3.7.17.1 (which is not yet implemented) states "the calculations and verification of acceptable configurations must be performed by a QA software program (i.e., the Surveillance Report cannot be performed manually or generated by a non-QA tool or spreadsheet)." If the PETRIFIED software were to fail, this procedural requirement will result in the inability to perform Region II surveillance activities (and therefore no fuel movement into or within Region II) until the issue is corrected and QA testing of any software changes are completed.

The TS limits which are programmed into PETRIFIED do not represent the current CPNPP storage limitations, and, therefore, the PETRIFIED software has not been implemented. To compensate for a lack of operational experience, the licensee submitted Commitment No. 4844167 in its letter dated May 1, 2014 (Reference 8). This commitment is intended to ensure TS 3.7.17 compliance while providing independent verifications of the PETRIFIED software with operational experience from two refueling cycles. The licensee stated that it will

perform the operational verification described in the paragraph below for all Region II fuel movement in 2014, which is expected to involve approximately 700-800 fuel movement steps.

Following implementation of the PETRIFIED software, the licensee will operationally verify (independent of PETRIFIED calculations) the TS 3.7.17 categorization calculations for fuel discharged from Unit 2 Cycle 14 and Unit 1 Cycle 17 prior to storing these assemblies in Region II. The licensee will further operationally verify (independent of PETRIFIED) fuel movement plans which impact Region II during 2014, to ensure the allowed configurations of TS 3.7.17 are maintained. Fuel movement plans which impact Region II includes movement of fuel into Region II as well as fuel movement within Region II (reconfiguration). These operational verification activities will be performed by an individual independent of the software developer and the developer of the initial PETRIFIED test plan, but maintains at least the same level of qualification.

Based on the software QA processes and the administrative controls implemented by the licensee to ensure that the proposed software functions as required and that errors are minimized, and based on the NRC staff's understanding that additional independent verification will be performed on the software for operational experience assessment, the staff concludes that the licensee's use of the PETRIFIED software is acceptable.

3.3.7 Potential for Spent Fuel Assembly Multiple Misload Event

On February 17, 2012, at Indian Point Unit 2, during evaluation of the effects of a previous forced maintenance outage on fuel characterization for storage, the Fuel Transfer Form preparer using TS 3.7.13, "Spent Fuel Pit Storage," as a reference, recognized an error that had allowed 11 fresh fuel assemblies to be moved into the SFP in a configuration not permitted by the TS (Reference 40). The direct cause for the error was due to the move sheets issued being incorrect. The error was a result of poor self and peer check/review during preparation and verification of the move sheets.

Hence, the NRC staff reviewed the application for the potential for multiple misloads and the administrative and procedural controls proposed by the licensee to minimize the potential for multiple fuel assembly misloads.

The NRC staff evaluated the configuration control software (CCS) proposed by the licensee to help reduce the effects of complex spent fuel storage configurations in Region II of the SFP and to reduce the potential for multiple fuel assembly misloads. The NRC staff concluded that the new CCS will function as a redundant (actually diverse, eliminating the common-cause potential[1]) enhancement to the existing fuel loading program. Enclosure 1 to the March 28, 2013, LAR describes features of the CCS, as follows:

- Categorizes each fuel assembly by category (via commercial ShuffleWorks [used throughout the nuclear industry] database, using data from CPNPP's TARPIT software), assigning to one of six classes. Since no input files need be created

[1] The one common-cause aspect that cannot be eliminated, although it can be reduced, is the potential for human error despite of all the controls and independent reviews. Even knowing where to load an assembly does not prevent a misload via human error.

(uses TARPIT directly), there is no "transcription" error potential. Graphical comparisons are developed to aid the user.

- Determines acceptability of SFP configuration (again using ShuffleWorks) in light of TS 3.7.16, providing color-coded SFP graphics to aid the user.

- Reviews ShuffleWorks sequence files on a move-by-move basis to identify any potential violations, providing a report determining acceptability of each movement and any reasons for failure.

- Highlights failure configurations, including but not limited to (a) fuel assembly stored in configuration not permitted by TS 3.7.17, even if just temporarily; (b) failure to satisfy interface requirements (configuration specific); (c) missing fuel information from database; (d) any input parameter outside the analyzed range; (e) multiple items stored in the same location, an impossibility, even temporarily; (f) storage in invalid locations, e.g., non-existent or oversized; (g) moving a fuel assembly into a "Restricted" region based on fuel parameter database; (h) storage of non-fuel items (e.g., trash basket) in cells assumed to be filled only with water.

- Generates proposed reports to support SR 3.1.17.1, including frequency of "Prior to moving a fuel assembly into any 'Restricted' storage location." Report provides information to support documentation of compliance with TS, enabling QA control and traceability.

- CCS program will be controlled per CPNPP software QA program, including (a) independent testing of software features for accuracy, completeness, reliability, functionality and ease of use; (b) independent review for even routing changes/updates; (c) independent testing of any changes.

Also, based on the operating experience at CPNPP and the proposed CCS program, the NRC staff determined that the probability of a multiple fuel assembly misload is extremely low and would be considered a very unlikely event.

Based on the above, the NRC staff concludes that the CCS features will significantly aid the operator in rendering the likelihood of a fuel misload negligible and that the combined effect of the current program and the CCS addition should significantly reduce the likelihood of human error.

3.3.8 Summary

The licensee submitted the LAR to address the non-conservatisms in the SFP NCS analysis of record and the associated TSs (Reference 1). The LAR is supported by Westinghouse Report, WCAP-17728-P, Revision 0, which documents the criticality analysis for CPNPP spent fuel storage (Reference 20). The proposed changes to TS 3.7.16, "Fuel Storage Pool Boron Concentration," TS 3.7.17, "Spent Fuel Assembly Storage," TS 4.3, "Fuel Storage," and TS 5.5, "Programs and Manuals," impose the storage requirements reflecting the new SFP criticality analysis.

The NRC staff reviewed the NCS analysis to ensure that the assumptions and analytical techniques used are adequately substantiated to conclude at a 95 percent probability, 95 percent confidence level, that the regulatory requirements will be met.

Based on its review of the licensee's Neutron Absorber Monitoring Program described in Section 3.3.4, the NRC staff concludes that the Boral neutron absorber will be adequately monitored for use in the SFP. Also, the staff concludes that the proposed Neutron Absorber Monitoring Program, which includes physical and neutron attenuation testing, is capable of detecting potential degradation of the Boral material that could impair its neutron absorption capability. Therefore, the staff concludes that the use of Boral as a neutron absorber panel in the spent fuel racks for subcriticality credit is acceptable with the use of the Neutron Absorber Monitoring Program.

Also, based on the review of the training program described in Section 3.3.5, software QA processes described in Section 3.3.6, and additional administrative controls proposed by the licensee and the potential for multiple fuel assembly misloads described in Section 3.3.7, the NRC staff concludes that the licensee's use of the PETRIFIED software and additional administrative controls is acceptable.

The licensee has shown compliance with the requirements of 10 CFR 50.36, 10 CFR 50.68, and 10 CFR Part 50, Appendix A, Criterion 62. The NRC also, in general, followed the guidance provided by the NRC Kopp letter, ISG DSS-ISG-2010-01, and SRP Sections 9.1.1 and 9.1.2.

Based on the above, the NRC staff concludes that there is reasonable assurance that CPNPP will comply with the applicable regulatory requirements specified in Section 2.0, "Regulatory Evaluation." Therefore, the NRC staff concludes that the proposed TS changes meet the requirements of 10 CFR 50.36,, 10 CFR 50.68, and Criterion 62 of 10 CFR 50, Appendix A and are acceptable.

4.0 REGULATORY COMMITMENTS

The licensee made the following regulatory commitments in its letters dated November 26 and December 17, 2013, and May 1, 2014 (References 4, 5, and 8).

Commitment No.	Description of the Regulatory Commitment	Source
4753383	The BORAL Monitoring Program ensures that: (a) future coupons removed for testing are reinserted into the SFP after testing is complete, (b) location and move times are tracked for the samples (to enable demonstration of the total duration that test coupons were removed from the pool environment), and (c) includes a requirement to perform continual testing every 10 years as long as the Region I storage racks are licensed to store fuel.	Reference 4

Commitment No.	Description of the Regulatory Commitment	Source
4753402	BORAL is credited in the Region I racks for both the borated and non-borated cases. The BACKGROUND section of the TS Bases B 3.7.16, 4th paragraph, 1st sentence will be changed to state: "In order to maintain keff less than or equal to 0.95, the presence of fuel pool soluble boron is credited for the storage of fuel assemblies within the Region I and Region II racks (in addition to the BORAL neutron absorber material in Region I)."	Reference 4
4753403	The supporting Criticality Safety Analysis demonstrated K_{eff} remains LESS THAN 1.0 for all analyzed conditions (reference WCAP-17728-P Rev 1, Section 2.1.1). The BACKGROUND section of the TS Bases B 3.7.16, 3rd paragraph, last sentence will be changed to state: "The neutron absorber material BORAL is credited for the storage of spent fuel assemblies within the Region I racks to maintain K_{eff} less than 1.0 at 0 ppm soluble boron concentration."	Reference 4
4760732	Current and future fuel assemblies will use a Wet Annular Burnable Absorber (WABA) which is no more than 120 inches in length and 95/95 Upper Bound fuel theoretical density (%TD) will be ≤ 96.5%. Future fuel cycles will utilize a Maximum Inlet Temperature of ≤ 561.7 degrees F. If these parameters are not met, the fuel assemblies will be treated as fresh fuel for storage in the Spent Fuel Pools.	Reference 5
4760737	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 0.194 atom fraction, 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.	Reference 5
4760738	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 0.194, a B-10 measurement will be performed on the Spent Fuel Pool prior to the next refueling outage to ensure the B-10 value in the SFP has not significantly changed.	Reference 5

Commitment No.	Description of the Regulatory Commitment	Source
4760741	For fuel assemblies which are classified as outlier assemblies solely due to Hot Full Power (HFP) Rodded Operation, 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. 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 area of applicability.	Reference 5
4844167	Following implementation of the PETRIFIED software, Luminant Power will operationally verify (independent of PETRIFIED calculations) the TS 3.7.17 categorization calculations for fuel discharged from Unit 2 Cycle 14 and Unit 1 Cycle 17 prior to storing these assemblies in Region II. CPNPP will further operationally verify (independent of PETRIFIED) fuel movement plans which impact Region II during 2014, to ensure the allowed configurations of TS 3.7.17 are maintained. Fuel movement plans which impact Region II includes movement of fuel into Region II as well as fuel movement within Region II (reconfiguration). These operational verification activities will be performed by an individual independent of the software developer and the developer of the initial PETRIFIED test plan, but maintains at least the same level of qualification.	Reference 8

The regulatory commitments described above are in support of the proposed TS changes and provided added assurance for the licensee's actions.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding

published in the *Federal Register* on November 5, 2013 (78 FR 66391). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446 - License Amendment Request (LAR) 13-01, Revision to Technical Specifications 3.7.16, "Fuel Storage Pool Boron Concentration," 3.7.17, "Spent Fuel Assembly Storage," 4.3, "Fuel Storage," and 5.5, "Programs and Manuals," dated March 28, 2013 (ADAMS Accession No. ML130950023).

2. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446 – Supplemental Information Supporting LAR 13-01 Spent Fuel Pool Criticality Analysis," dated July 16, 2013 (ADAMS Accession No. ML13205A056).

3. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446 - License Amendment Request (LAR) 13-01, Supplement Spent Fuel Pool Criticality Analysis Removal Technical Specifications 3.7.16, "Fuel Storage Pool Boron Concentration," 3.7.17, "Spent Fuel Assembly Storage," 4.3, "Fuel Storage," and 5.5, "Programs and Manuals," dated October 22, 2013 (ADAMS Accession No. ML13309A026).

4. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446 - License Amendment Request (LAR) 13-01, Supplement Spent Fuel Pool Criticality Analysis Removal Technical Specifications 3.7.16, "Fuel Storage Pool Boron Concentration," 3.7.17, "Spent Fuel Assembly Storage," 4.3, "Fuel Storage," and 5.5, "Programs and Manuals," dated November 26, 2013 (ADAMS Accession No. ML13346A175).

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7. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446 – License Amendment Request (LAR) 13-01, Spent Fuel Pool Criticality Analysis, Response to Requests for Additional Information," dated April 17, 2014 (ADAMS Accession No. ML14118A094).

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16. Singal, B. K., U.S. Nuclear Regulatory Commission, letter to Mike Blevins, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 - Issuance of Amendments [146] Re: License Amendment Request 07-004, Revision to Operating License and Technical Specification 1.0, "Use and Application," to Revise Rated Thermal Power From 3458 MWt to 3612 MWt, dated June 27, 2008 (ADAMS Accession No. ML081510173).

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Date: July 1, 2014

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 162 to NPF-87
2. Amendment No. 162 to NPF-89
3. Safety Evaluation (non-proprietary)
4. Safety Evaluation (proprietary)

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