April 2, 2008

Mr. M. R. Blevins
Executive Vice President
& Chief Nuclear Officer
Luminant Generation Company LLC
ATTN: Regulatory Affairs
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATIONS TO ALLOW USE OF WESTINGHOUSE-DEVELOPED/NRC-APPROVED ANALYTICAL METHODS TO ESTABLISH CORE OPERATING LIMITS (TAC NOS. MD5243, MD5244, MD6212, AND MD6213)

Dear Mr. Blevins:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 144 to Facility Operating License No. NPF-87 and Amendment No. 144 to Facility Operating License No. NPF-89 for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 10, 2007, as supplemented by letters dated July 31, August 16, November 15 (two letters), and November 19, 2007, and February 11, March 6, March 13, and March 26, 2008.

The amendments revise CPSES, Units 1 and 2, TSs 3.1, "Reactivity Control Systems," TS 3.2, "Power Distribution Limits," TS 3.3, "Instrumentation," and TS 5.6.5b, "Core Operating Limits Report (COLR)," to incorporate standard Westinghouse-developed and NRC-approved analytical methods into the list of methodologies used to establish the core operating limits.

A copy of our related Safety Evaluation is enclosed. 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
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 144 to NPF-87
2. Amendment No. 144 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession Nos.: Pkg ML080500666 (Amendment ML080500627, License/TS Pgs ML080510083)

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DSS/SRXB/BC	DSS/SNPB/BC	DIRS/ITSB/BC	OGC	NRR/LPL4/BC
NAME	BSingal	JBurkhardt	GCranston (*)	AMendiola(**)	GWaig(**)	MSmith, NLO	THiltz
DATE	3/26/08	3/26/08	1/21/08	3/4/08	3/7/08	3/28/08	4/2/08

OFFICIAL AGENCY RECORD (*) SE input memo (**) See previous concurrence

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LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 144
License No. NPF-87

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

   A. The application for amendment by Luminant Generation Company LLC dated April 10, 2007, as supplemented by letters dated July 31, August 16, November 15 (two letters), and November 19, 2007, and February 11, March 6, March 13, and March 26, 2008, 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. 144 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 as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to startup from refueling outage 10 for Comanche Peak Steam Electric Station, Unit 2.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
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: April 2, 2008

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 144
License No. NPF-89

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

   A. The application for amendment by Luminant Generation Company LLC dated April 10, 2007, as supplemented by letters dated July 31, August 16, November 15 (two letters), and November 19, 2007, and February 11, March 6, March 13, and March 26, 2008, 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. 144 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.

4. The license amendment is effective as of its date of issuance and shall be implemented prior to startup from refueling outage 10 for Comanche Peak Steam Electric Station, Unit 2.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
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: April 2, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 144

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 144

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	REMOVE	INSERT
i	i	3.2-11	3.2-16
iv	iv	--	3.2-17
3.1-16	3.1-16	3.2-12	3.2-18
3.1-17	3.1-17	3.2.13	3.2-19
3.2-1	3.2-1	3.2-14	3.2-20
3.2-2	3.2-2	3.2-15	3.2-21
3.2-3	3.2-3	3.3-10	3.3-10
3.2-4	3.2-4	3.3-11	3.3-11
3.2-5	3.2-5	5.0-33	5.0-33
--	3.2-6	5.0-34	5.0-34
--	3.2-7	--	5.0-35
--	3.2-8	5.0-35	5.0-36
--	3.2-9	5.0-36	5.0-37
--	3.2-10	5.0-36a	5.0-38
3.2-6	3.2-11	5.0-36b	5.0-39
3.2-7	3.2-12	5.0-37	5.0-40
3.2-8	3.2-13	5.0-38	5.0-41
3.2-9	3.2-14	5.0-39	5.0-42
3.2-10	3.2-15	5.0-40	5.0-43
		5.0-41	5.0-44

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 144 TO

FACILITY OPERATING LICENSE NO. NPF-89

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By letter dated April 10, 2007 (Reference 1) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071070307), and supplemented by letters dated July 31 (Reference 2), August 16 (Reference 17), November 15 (Reference 10), November 15 (Reference 18, Non-Publicly Available) (two letters), and November 19, 2007 (Reference 11), and February 11 (Reference 40), March 6 (Reference 41), March 13, 2008 (Reference 42), and March 26, 2008 (Reference 43) (ADAMS Accession Nos. ML072200641, ML072340228, ML073241095, ML073241097, ML073330059, ML080520043, ML080710563, ML080840547, and MLxxxxxxxxx[1], respectively), TXU Generation Company LP (subsequently renamed as Luminant Generation Company LLC) (the licensee), submitted changes to the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, Technical Specifications (TSs) to allow the use of several Westinghouse-developed and U.S. Nuclear Regulatory Commission (NRC)-approved analytical methods to establish core operating limits that are documented in the CPSES, Units 1 and 2 Core Operating Limit Report (COLR).

The supplemental letters dated July 31, August 16, November 15 (two letters), and November 19, 2007, and February 11, March 6, March 13, and March 26, 2008, 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 August 14, 2007 (72 FR 45461).

For previous cycles, the licensee has been performing their own loss-of-coolant accident (LOCA) and non-LOCA transient analyses based on their own methods to support CPSES, Units 1 and 2 operation. For future cycles beginning with CPSES, Unit 2 Cycle 11 (spring 2008), the licensee has elected to apply the Westinghouse methodologies to perform LOCA and non-LOCA

1 As of the date of issuance of these amendments, document has not yet been placed into ADAMS.

transient analyses. In the process of converting to Westinghouse safety analysis methodologies, the licensee proposes changes in the following areas: (1) implementation of Westinghouse methodologies for determining selected core operating parameter values; (2) implementation of relaxed axial offset control (RAOC) of the reactor core; and (3) implementation of the BEACON method for determining the core power distribution. The affected TS sections are TS 3.1, "Reactivity Control Systems," TS 3.2, "Power Distribution Limits," TS 3.3, "Instrumentation," and TS 5.6.5b, "Core Operating Limits Report (COLR)."

CPSES, Units 1 and 2, Nuclear Steam Supply System (NSSS) is a Westinghouse 4-loop design, currently operating at 3458 megawatts thermal (MWt) (1.4 percent above originally licensed thermal power), operating predominately with 17x17 VANTAGE+ fuel assemblies and a few Optimized Fuel Assemblies. The CPSES, Unit 2 license amendment is planned for implementation prior to startup from the spring 2008 refueling outage and the CPSES, Unit 1 amendment is planned for implementation prior to startup from the fall 2008 refueling outage.

By letter dated July 31, 2007 (Reference 2), as supplemented by letters dated November 15 and 19, 2007 (References 10 and 11), the licensee submitted the evaluation models and results for the large break LOCA (LBLOCA) analyses for CPSES, Units 1 and 2 performed using the Westinghouse Automated Statistical Treatment of Uncertainty Method (ASTRUM) Best-Estimate Large Break LOCA (BE-LBLOCA) methodology, and the CPSES, Units 1 and 2 evaluation models for the Westinghouse NOTRUMP-based small break LOCA (SBLOCA) analyses. This safety evaluation (SE) provides an evaluation of application of the Westinghouse BE-LBLOCA (ASTRUM) and SBLOCA (NOTRUMP) methodologies to CPSES Units 1 and 2.

Westinghouse obtained generic NRC approval of its original topical report (TR) describing the BE-LBLOCA methodology in 1996 for 3- and 4-loop pressurized-water reactors (PWRs). CPSES units are 4-loop PWRs. NRC approval of the methodology is documented in the NRC safety evaluation report (SER) appended to the TR (Reference 3). This methodology was later extended to 2-loop Westinghouse plants with Upper Plenum Injection (UPI) in 1999 as documented in the NRC SER appended to the UPI TR (Reference 4).

Westinghouse recently completed a program to revise the statistical approach used to develop Peak Cladding Temperature (PCT) and oxidation results at the 95th percentile. This method is based on the Code Qualification Document (CQD) methodology (References 3 and 4) and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology (NUREG/CR-5249). However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method in which the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A (Reference 5).

The SBLOCA analyses for CPSES, Units 1 and 2 were completed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) (References 6, 7, and 8). The NOTRUMP-EM SBLOCA analyses were performed at 100.6 percent of the Stretch Power Uprate (SPU) core power of 3612 MWt (NSSS power of 3628 MWt) for both CPSES units. The CPSES, Unit 1 LOCA models included the Δ-76 Replacement Steam Generator (RSG).

The purpose of the licensee's analysis was to demonstrate conformance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46 (Reference 9) requirements at the planned SPU conditions with the ASTRUM and the NOTRUMP-EM. Important input assumptions, as well as analytical models and analysis methodology for the BE-LBLOCA and the SBLOCA were provided. Analysis results were also provided which showed no design or regulatory limit related to the BE-LBLOCA and the SBLOCA would be exceeded at the conditions analyzed.

In its July 31 and November 15 submittals (References 2 and 10), the licensee stated that both the licensee and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses, and ensure that the values and ranges of LBLOCA and SBLOCA analysis inputs for peak cladding temperature and oxidation-sensitive parameters bound the ranges and values of the as-operated plant parameters. The NRC staff finds that this statement, along with the generic acceptance of ASTRUM and NOTRUMP methodologies, provides reasonable assurance that ASTRUM and its BE-LBLOCA analyses, and NOTRUMP and its SBLOCA analyses apply to CPSES, Units 1 and 2 operated at the proposed SPU. Furthermore, these interface processes, along with vendor internal processes for assessing evaluation model changes and errors, are used to identify the need for LOCA analyses impact assessments.

2.0 REGULATORY EVALUATION

The LBLOCA and SBLOCA analyses were performed to demonstrate that the system design would provide sufficient Emergency Core Cooling System (ECCS) flow to transfer the heat from the reactor core following a LOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than enough to compromise cladding ductility and would not result in excessive hydrogen generation. The NRC staff reviewed the analyses to assure that the analyses reflected suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities were available such that the safety functions could be accomplished, assuming a single failure, for LOCAs considering the availability of onsite and offsite electric power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available, with offsite electric power available). The acceptance criteria for ECCS performance are provided in 10 CFR 50.46, and were used by the NRC staff in assessing the acceptability of the Westinghouse ASTRUM and NOTRUMP methodologies for CPSES, Units 1 and 2.

The NRC staff also reviewed the limitations and conditions stated in its SE supporting approval of the Westinghouse ASTRUM and NOTRUMP methodologies and the range of parameters described in the ASTRUM and NOTRUMP TRs in its assessment of the acceptability of the methodology for CPSES, Units 1 and 2.

The NRC staff reviewed the amendment request to evaluate the applicability of the Westinghouse non-LOCA methodologies to CPSES, Units 1 and 2, confirm that the use of the methodologies is within the NRC-approved ranges, and verify that the results of non-LOCA analyses are in compliance with the requirements of General Design Criteria (GDC) 10 and GDC 15.

GDC 10 of Appendix A to 10 CFR, Part 50 requires that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrence (AOO). GDC 15 requires that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during normal operation, including AOOs.

CPSES, Units 1 and 2 NSSS is a Westinghouse 4-loop design. The NRC staff utilized NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," in its review of the proposed TS changes for CPSES, Units 1 and 2.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's request to evaluate its applicability to the LOCA and non-LOCA analyses and the results of the evaluation are discussed separately for LOCA and non-LOCA analyses.

3.1 LOCA Analyses

In general, methodologies or computer codes used to support licensing applications involving LOCA analyses are documented in TRs, which are reviewed by the NRC staff on a generic basis. In the NRC staff SE approving the TR, the NRC staff defines the basis for acceptance in conjunction with any limitations and conditions on the applications as appropriate. The limitations and conditions on the licensee's use of these TRs require a demonstration that they are compatible with the CPSES's LOCA analysis.

3.1.1 Best-Estimate LBLOCA (BE-LBLOCA)

NRC Staff Evaluation

A major rupture (large break) is defined as a breach in the RCPB with a total cross-sectional area greater than 1.0 square foot. The analysis was performed in compliance with all the NRC conditions and limitations, as identified in WCAP-16009 (Reference 5).

In its submittal the licensee provided the results, including the sequence of events for the limiting PCT transient, for the CPSES, Units 1 and 2 BE-LBLOCA analyses at 3612 MWt (+0.6 percent Core Power Calorimetric Uncertainty (CPCU)) performed in accordance with the ASTRUM methodology. The major plant parameter assumptions used in the BE-LBLOCA analyses for CPSES, Units 1 and 2 were also provided in the submittal. The licensee's limiting results (which occur in Unit 2) for the calculated PCT, the local maximum oxidation (LMO) for cladding, and the core wide oxidation (CWO) for cladding are provided in the Table 1 along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 1: Limiting LBLOCA Analysis Results (Occurs in Unit 2)

Parameter	ASTRUM Results	10 CFR 50.46 Limits
Limiting Break Size/Location	DEG[1]/PD	N/A
Peak Clad Temperature	1632 °F	2200 °F (10 CFR 50.46(b)(1))
Local Maximum Oxidation	0.71%	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	0.00%	1.0% (10 CFR 50.46(b)(3))

[1] DEG/PD is a double-ended guillotine break at the pump discharge.

The results presented in Reference 2 show the effect of the effective break area on the analyzed PCT. The effective break area is calculated by multiplying the discharge coefficient with the sample value of the break area, normalized to the cold-leg cross sectional area. The results show that the break area is a significant contributor to the variation in PCT. Due to relatively low PCT results, CWO remained negligible for all cases. The worst single failure for the CPSES BE-LBLOCA analysis is the loss of one train of ECCS injection (consistent with the ASTRUM Topical). All containment systems, which would reduce containment pressure, were modeled for the BE-LBLOCA containment backpressure calculation.

The NRC staff concluded that the results of the BE-LBLOCA analyses for CPSES, Units 1 and 2 (Table 1) demonstrated compliance with 10 CFR 50.46(b)(1) through (b)(3) for licensed power levels of up to 3612 MWt. Meeting these criteria provides reasonable assurance that at the SPU level at CPSES, Units 1 and 2, the core will be amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of CPSES, Units 1 and 2 to satisfy the long-term cooling requirements of 10 CFR 50.46(b)(5) is addressed in Section 3.1.3.

Results of the NRC Staff Evaluation

Based on its review as discussed above, the NRC staff concluded that the Westinghouse ASTRUM methodology, as described in WCAP-16009-P-A, is acceptable for use for CPSES, Units 1 and 2 in demonstrating compliance with the requirements of 10 CFR 50.46(b). The NRC staff's conclusion was based on the assumed SPU core power up to 3612 MWt (+0.6 percent CPCU).

The NRC staff's review of the acceptability of the ASTRUM methodology for CPSES, Units 1 and 2 focused on assuring that the CPSES, Units 1 and 2 specific input parameters or bounding values and ranges (where appropriate) were used to conduct the analyses, that the analyses were conducted within the conditions and limitations of the NRC-approved Westinghouse ASTRUM methodology, and that the results satisfied the requirement of 10 CFR 50.46(b) based on a licensed power level of up to 3612 MWt.

This SE documents the NRC Staff review and the bases of acceptance of the Westinghouse ASTRUM BE-LBLOCA analysis methodology for application to the CPSES, Units 1 and 2 and of the LBLOCA analyses discussed above which were performed with the ASTRUM methodology for reference in the SPU of CPSES, Units 1 and 2.

3.1.2 Small-Break LOCA (SBLOCA)

NRC Staff Evaluation

The SBLOCA includes all postulated pipe ruptures with a total cross-sectional area less than 1.0 square foot. The SBLOCAs analyzed in this section are for those breaks beyond the capability of a single charging pump resulting in the actuation of the ECCS. The analysis was performed to demonstrate conformance with the 10 CFR 50.46 requirements for the conditions associated with CPSES, Units 1 and 2.

In areas where the licensee and its contractors used NRC-approved methods in performing analyses, the NRC staff reviewed relevant material to assure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the effects of the changes in plant operating conditions on the use of these methods to assure that the methods were appropriate for use at the proposed SPU conditions.

The SBLOCA analyses for CPSES, Units 1 and 2 were performed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) as described in WCAP-10054-P-A, WCAP-10054-P-A, Addendum 2, and WCAP-10079-P-A (References 6, 7, and 8). The NOTRUMP-EM SBLOCA analyses were performed at 100.6 percent of the SPU core power of 3612 MWt (NSSS power of 3628 MWt) for both CPSES units. The CPSES, Unit 1 NOTRUMP model included the Δ-76 RSG. The purpose of this analysis is to demonstrate conformance with the 10 CFR 50.46 (Reference 9) requirements at the SPU conditions with NOTRUMP-EM. The analysis has shown that no design or regulatory limit related to the SBLOCA would be exceeded at the conditions analyzed.

The SBLOCA methodology using the NOTRUMP-EM was developed in accordance with the requirements of 10 CFR 50 Appendix K. This regulation was designed to produce a conservative prediction of the analysis results and includes various conservative modeling requirements such as the decay heat model (1971 ANS Infinite +20 percent), the zirconium-water reaction model (Baker-Just) and the most limiting single failure criterion. For the SBLOCA analysis, loss-of-offsite-power (LOOP) was assumed, which resulted in the limiting single failure assumption of the loss of one Emergency Diesel Generator (EDG) and a subsequent loss of one train of pumped ECCS. The SBLOCA analysis assumes that reactor trip occurs coincident with the LOOP, which results in the following: (a) Reactor Coolant Pump (RCP) trip and coastdown and (b) Steam Dump System being inoperable.

The SBLOCA analyses considered five different break cases each for CPSES, Units 1 and 2. A spectrum of cold leg breaks of equivalent diameters 2, 3, 4 and 6 inches and an accumulator line break of equivalent diameter 8.75 inches were considered. For both CPSES Units 1 and 2, the 4-inch break was found to be limiting for PCT. For both units the 2-, 6-, and 8.75-inch breaks resulted in minimal or no core uncovery and therefore PCT information was not calculated for these breaks. The intermediate (non-integer) breaks were not considered because the limiting PCTs for the 4-inch break have considerable margin to the 2200 degrees Fahrenheit (°F) limit set forth by the acceptance criteria for the SBLOCA analysis.

When an SBLOCA occurs, depressurization of the reactor coolant system (RCS) causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1860 pounds per square inch absolute (psia), is reached. LOOP is postulated to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1715 psia is reached. Safety injection flow is delayed 22 seconds after the occurrence of the low-pressure condition. This delay accounts for signal processing, diesel generator start up and emergency power bus loading consistent with the LOOP coincident with reactor trip, as well as the pump acceleration and valve delays.

The peak cladding temperature is 1013 °F for CPSES, Unit 1 (4-inch break) and 1209 °F for CPSES, Unit 2 (4-inch break), and the maximum local transient oxidation is 0.02 percent for CPSES, Unit 1 and 0.05 percent for CPSES, Unit 2. The sum of the pre-transient and transient oxidation remains below 17 percent at all times in life and the average oxidation is negligible for both units. The licensee's limiting results (which occur in CPSES, Unit 2) for the calculated PCT, the maximum local oxidation for cladding, and the core wide oxidation for cladding are summarized in Table 2 along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 2: Limiting SBLOCA Analysis Results (Occurs in Unit 2)

Parameter	NOTRUMP Results	10 CFR 50.46 Limits
Limiting Break Size	4-inch	N/A
Peak Clad Temperature	1209 °F	2200 °F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	0.05%	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	Negligible	1.0% (10 CFR 50.46(b)(3))

Since the licensee's analysis was conducted consistent with its licensing basis and the NRC-approved methodology, and the significant margin to the 10 CFR 50.46(b)(1) 2200 °F PCT acceptance criteria, the NRC staff concluded that the plant response following an SBLOCA at SPU conditions is acceptable.

Results of the NRC Staff Evaluation

Based on the appropriate application by the licensee of NRC-approved methodologies to analyze SBLOCAs for CPSES, Units 1 and 2, the NRC staff concludes that operation of CPSES, Units 1 and 2 at SPU conditions is acceptable and both the CPSES units will be able to mitigate the consequences of SBLOCAs. Therefore, the NRC staff concludes it has reasonable assurance that for SBLOCAs, the acceptance criteria of 10 CFR 50.46(b)(1), (2), and (3) related to PCT, local oxidation, and hydrogen generation, respectively, are satisfied for CPSES, Units 1 and 2 at SPU conditions.

3.1.3 Post-LOCA Long-Term Cooling

NRC Staff Evaluation

The licensee's boric acid precipitation calculation model meets NRC guidance as presented in Reference 12 and was consistent with the interim methodology reported in Reference 13. The licensee stated that, consistent with the guidance provided in Reference 13, the Beaver Valley extended power uprate (EPU) approach (References 14 and 15) was used for the CPSES, Units 1 and 2 post-LOCA long-term cooling analysis.

The NRC staff evaluated the post-LOCA long term cooling analyses. The NRC staff's evaluation also included an audit calculations pertaining to post-LOCA long-term cooling, upon which certain accident analyses, presented in the application, were based. The NRC staff performed independent calculations to assess the timing for boric acid precipitation for both large and small breaks as discussed below.

The NRC staff reviewed the post-LOCA Long Term Cooling analyses presented in Section 2.7.3.4 of the CPSES, Units 1 and 2, Transition of Methods Safety Analyses Report (Reference 16). This evaluation deals with the prevention of boric acid precipitation following both large and small break LOCAs. The key results of this evaluation is the timing for switch to simultaneous injection and the timing for initiation of cooldown of the RCS to control boric acid for SBLOCAs where simultaneous injection may not be successful.

The limiting LBLOCA, analyzed by the licensee, showed that the precipitation time was calculated to be about 7.5 hours following opening of the break, without crediting the switch to simultaneous injection. The analysis included the following key assumptions:

- The 1971 ANS [American Nuclear Society] decay heat standard with a 20 percent increase,
- The mixing volume was computed as a function of time including loop pressure drop effects,
- The upper plenum pressure was assumed to be at the minimum value of 14.7 psia,
- The precipitation limit of 27.5 wt% was assumed,
- No credit was taken for containment pH (potential of hydrogen) additives that would increase the solubility limit,
- The void distribution was computed using the Yeh Correlation, and
- Core power was assumed to be 3628 MWt including a 2 percent uncertainty.

The NRC staff performed audit calculations and calculated the precipitation time of 4.9 hours. This calculation assumed that the break was located on the top of the discharge leg and emergency core coolant filled the suction piping. This increased and maximized the loop resistance during the long term due to the liquid head of water in the vertical section of the suction piping. The inclusion of water trapped in the loop seal piping produces a conservatively calculated precipitation time, since the mixing volume is minimized during the event due to the increased pressure drop in the loops. The increased pressure drop will depress the two-phase level deeper into the inner vessel containing the lower plenum, core and upper plenum.

The licensee conservatively chose the switch time for simultaneous injection to be at 3 hours post-LOCA. This timing is considered acceptable and provides at least 1 hour or more to complete the realignment. Based on the NRC staff calculations, the NRC staff finds the timing of 3 hours to switch to simultaneous injection to be acceptable, and includes the possibility for loop seal refilling and increased loop resistance during the LOCA.

The NRC staff also reviewed the results of the SBLOCA analyses which includes break sizes 0.1 square feet down to and including 0.001 square feet. Break sizes 0.1 to 0.005 square feet have a minimum RCS pressure of 120 psia that represents a very high solubility limit. Although the refill timing for these larger small breaks is delayed, the higher solubility limit will maintain the boric acid soluble until refill has been achieved. The analyses demonstrated that the RCS will refill and disperse boric acid prior to exhaustion of the condensate storage tank and is a result of the operator instruction to initiate a cooldown no later than 1 hour post-LOCA. Break sizes 0.005 square feet and smaller will show an early refill and re-establishment of natural circulation that disperses the boric acid throughout the RCS. Below 0.001 square feet the charging system will maintain the RCS in a subcooled fluid state.

Based on the review of the long term cooling evaluation and the audit calculations of the precipitation timing for the limiting LBLOCA, the NRC staff finds the long term cooling analysis acceptable at 3268 MWt and refueling water storage tank boron concentration of 2400 parts per million (ppm).

The licensee will revise the Updated Final Safety Analysis Report (UFSAR), TSs, and emergency operating procedures (EOPs) to support the maximum time to establish simultaneous hot leg and cold leg injection. ECCS recirculation flows are evaluated by comparing minimum safety injection pump flows to the flows necessary to dilute the core and replace core boil-off, thus keeping the core quenched.

Results of the NRC Staff Evaluation

The NRC staff considers the analyses and operator actions to be an acceptable approach for controlling boric acid precipitation for the CPSES, Units 1 and 2 NSSS at the proposed SPU operating conditions. Based on its review, the NRC staff finds the analyses, operator actions, and EOP changes to facilitate the successful control of boric acid following all LOCAs provides reasonable assurance that the long-term cooling requirements of 10 CFR 50.46(b)(5) are satisfied for CPSES, Units 1 and 2 under SPU conditions.

Conclusion (LOCA Analyses)

The NRC staff reviewed the Westinghouse ASTRUM BE-LBLOCA, NOTRUMP SBLOCA and the post-LOCA long-term cooling analyses for application to the CPSES, Units 1 and 2 NSSS operating under the proposed SPU conditions. The NRC staff's review confirmed that the licensee and its vendor have processes to assure that the CPSES-specific input parameter values and ranges and operator action times (where appropriate) that were used to conduct the analyses will assure that 10 CFR 50.46 limits are not exceeded, and that long-term cooling can be assured for all break sizes by providing the means to remove decay heat for extended periods, while also preventing the precipitation of boric acid for all break sizes and locations. Furthermore, the NRC staff finds that the analyses were conducted within the conditions and limitations of the NRC-approved Westinghouse ASTRUM BE-LOCA and NOTRUMP SBLOCA methodologies, and that the results satisfied the requirements of 10 CFR 50.46(b), based on the proposed SPU conditions. The NRC staff notes that the procedures for assuring boric acid control for all breaks for the CPSES, Units 1 and 2 NSSS are unique to this system and finds the vendor and licensee approach to be a conservative and acceptable approach for demonstrating core cooling during the long term for all break sizes.

3.1.4 Related Technical Specification Changes

TS 5.6.5b Core Operating Limits Report (COLR)

The licensee requested to modify TS 5.6.5b by revising the listing of approved analytical methods and the NRC staff found it to be acceptable. The licensee's proposed changes add the following NRC-approved analytical methods to TS 5.6.5b:

- WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.
- WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.
- WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.
- WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," April 1999.
- WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
- WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.
- WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.
- WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.
- WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3.2 Non-LOCA Analysis

In support of the LAR for conversion to Westinghouse methods for non-LOCA analyses, the licensee initially provided information in Reference 1 for the NRC staff to review. While Reference 1 provided a description of the proposed TS changes, the NRC staff found that sufficient information was not provided to justify the proposed TS changes. In particular no information was provided to justify the application of previously approved WCAP TRs specifically to CPSES, Units 1 and 2. No plant-specific analysis was provided to demonstrate the applicability of the TRs or show that the analysis acceptance criteria are met.

In general, methodologies or computer codes used to support licensing applications involving non-LOCA analyses are documented in TRs, which are reviewed by the NRC staff on a generic basis. In the NRC staff SE approving the TR, the NRC staff defines the basis for acceptance in conjunction with any limitations and conditions on the applications as appropriate. A generic TR describing a methodology or computer code does not provide the full justification for each plant-specific application. In situations where a plant-specific LAR references a TR that has not been previously applied, the NRC staff requests that the licensee submit a plant-specific analysis to demonstrate applicability of the TR.

The licensee subsequently submitted adequate information for the NRC staff to start the review (Reference 17). Specifically, the licensee performed analyses of the UFSAR events based on the Westinghouse methods to demonstrate that the use of the methods is within the approved ranges and that the results of the analyses meet the applicable acceptance criteria. In addition, the licensee provided information indicating compliance with the NRC staff SE limitations and conditions imposed for application of the TRs.

The NRC staff has reviewed the LAR (Reference 1) in conjunction with the supplemental information provided (Reference 17) and the responses to the request for additional information (RAI) (Reference 18) to evaluate the acceptability of use of the Westinghouse non-LOCA methodologies for CPSES, Units 1 and 2 licensing applications and to confirm adequate technical basis for the proposed TS changes. The NRC staff SE is provided in subsequent sections; Section 3.2.1 discusses the applicability of Westinghouse codes and methods, Section 3.2.2 discusses the limiting UFSAR transient analyses, and Section 3.2.3 discusses the TS changes.

3.2.1 Applicability of Westinghouse Codes and Methods

3.2.1.1 Overpower ΔT and Overtemperature ΔT Trip Functions (WCAP-8745-P-A)

The overpower ΔT [change in temperature] and overtemperature ΔT trip functions are designed to provide protection against fuel centerline melt (overpower ΔT) and departure from nucleate boiling (DNB) (overtemperature ΔT) during anticipated operational occurrences. The licensee proposes to use the methodology documented in Reference 19, WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Overtemperature ΔT Trip Functions," for determination of the ΔT trip functions. The parameter values of the ΔT trip functions are determined based on the following design criteria: (1) the uranium dioxide melting temperature is not exceeded for 95 percent of the fuel rods at the 95 percent confidence level; and (2) at least a 95 percent probability that DNB does not occur at the limiting fuel rod at a 95 percent confidence level. To satisfy these design acceptance criteria, the licensee proposed to limit the calculated fuel centerline temperature to less than 4700 °F, and limit the minimum calculated departure from nucleate boiling ratio (DNBR) no less than the correlation limit (1.17 for the WRB-2 DNBR correlation). The NRC staff finds that the proposed fuel centerline temperature limit is consistent with that specified in the approved Westinghouse TR, WCAP-8745-P-A (Reference 19), and the DNB correlation limit was previously approved by NRC for the 17x17 VANTAGE+ fuel used in the CPSES, Units 1 and 2 (References 20 and 21).

Trip system at CPSES, Units 1 and 2 is based on N-16 (Nitrogen-16). The licensee states that even though the Reference 19 methodology was developed for the generic Westinghouse overtemperature and overpower ΔT core protection systems, the protection philosophy is applicable to the overtemperature and overpower N-16-based core protection system used at CPSES, Units 1 and 2. The licensee states that the overtemperature and overpower systems are functionally equivalent; the ΔT-based system uses the hot-leg-to-cold-leg temperature difference as an indication of the reactor power, whereas the N-16 based system uses the normalized N-16 gamma activity measured in the hot leg as an indication of the reactor power. Both of the overtemperature systems serve as primary protection functions for the prevention of conditions that could result in DNB, and both of the overpower systems serve as primary protection functions for the prevention of conditions that could result in exceeding the linear heat generation rate (LHGR) limits.

The NRC staff previously reviewed and approved the implementation of N-16 trip model in the licensee-developed RETRAN code in Reference 22. In response to an NRC staff RAI, the licensee stated that the N-16 trip model, including time constants and trip response times, in the Westinghouse RETRAN code remains unchanged from that in the licensee-developed RETRAN code, which was accepted by the NRC staff in Reference 22. Based on above, the NRC staff has reasonable assurance that the N-16 modeling is acceptable in the Westinghouse RETRAN code.

The licensee used the ΔT trip model and recalculated the reactor trip setpoints as previously approved by References 19 and 22. The adequacy of these setpoints was confirmed by showing that the DNB design basis is met in the non-LOCA analyses (Section 3.2.2) that credit these functions for mitigation. Therefore, the NRC staff concluded that the licensee's use of the methodologies discussed in WCAP-8745-P-A is acceptable.

3.2.1.2 Relaxation of Constant Axial Offset Control (WCAP-10216-P-A)

The current procedure based on constant axial offset control (CAOC), for axial power distribution control requires that axial offset be kept within a narrow band about a target value during normal plant operation in order to ensure that unacceptable power shapes do not occur. The licensee proposed to use the relaxation of constant axial offset control (RAOC) procedure documented in WCAP-10216-P-A (Reference 40), "Relaxation of Constant Axial Offset Control/FQ Surveillance Technical Specification," to increase operating flexibility. The result of the RAOC procedure is a curve of allowed axial flux difference, ΔI % (difference between the upper and lower excore detector readings), as a function of power.

The RAOC procedure requires that the allowed axial flux difference, ΔI %, be bounded by those used in the analysis for LOCA events and non-LOCA Condition II events. With the RAOC implemented, TS must be established to require that the axial flux difference be maintained within the acceptable band as a function of power. The acceptability of the associated TS changes is discussed in Section 3.2.3 of this SE.

WCAP-10216-P-A is comprised of two parts. Part A covers the Relaxation of Constant Axial Offset Control. Part B covers the F_Q Surveillance Technical Specification. The licensee had previously implemented the Part B; however, implementing Part A affects the implementation of Part B. The NRC staff has become aware of a change in methodology for the F_Q Surveillance Technical Specification portion of WCAP-10216-P-A (Reference 40). The change in methodology is associated with occurrences when the measured axial offset differs from the predicted axial offset. The issue arises in the surveillance intended to determine whether the F_Q limit would be met during the non-equilibrium conditions following a plant maneuver. Based on an audit conducted in January 2008, the NRC staff determined that a change in methodology was being proposed for use at CPSES, Units 1 and 2. In response to the NRC staff RAI, in Reference 41, the licensee proposed a site-specific method for addressing occurrences when the measured axial offset differs from the predicted axial offset.

To determine whether the $F_Q(z)$ limit would be met during the non-equilibrium conditions following a plant maneuver, the measured $F_Q(z)$ is multiplied by the factor W(z). The equation for W(z) is shown on page 1 of Reference 41. For the purposes of the licensee's analysis, W(z) may be described as the ratio of the maximum predicted $F_Q(z)$ during the non-equilibrium conditions following a plant maneuver to the predicted $F_Q(z)$ at the predicted steady-state axial power shape P(z). If the actual steady-state P(z) differs from the predicted steady-state P(z), the pre-determined W(z) acquires some ambiguity. To remove the ambiguity, the licensee proposes using a ratio of predicted steady-state axial P(z) to measured steady-state P(z) to evaluate the impact on the margin to the $F_Q(z)$ limit, equations 1 and 2 of Reference 41. The intent is to isolate the effect of radial power peaking ($F_{xy}(z)$) on $F_Q(z)$, since $F_Q(z)$ may be represented as $P(z)*F_{xy}(z)$. The licensee's approach seems reasonable. However, it appears to be dependent on the unstated assumption that the maximum predicted $F_Q(z)$ during the non-equilibrium conditions following a plant maneuver will always exceed the measured $F_Q(z)$. Should the ratio of those two values approach 1.0, the loss of margin may not be recognized. Should the ratio drop below 1.0, the $F_Q(z)$ limit may not be met during the non-equilibrium conditions following a plant maneuver. Therefore, the staff believes it is prudent to place a limitation on that ratio for the licensee's proposal, that the ratio always exceeds 1.07. The staff believes a limitation that the value W(z) * [Predicted P(z) / Measured P(z)] is not less than 1.04 is reasonable to ensure the unstated assumption remains valid without undue burden on the licensee.

Therefore, the staff believes the licensee's site-specific proposal evaluating the F_Q surveillance for whether the $F_Q(z)$ limit would be met during the non-equilibrium conditions following a plant maneuver, when the actual steady-state P(z) differs from the predicted steady-state P(z) is acceptable. As indicated in the licensee's regulatory commitment in Reference 43, the licensee commits to the following: .

> Luminant Generation Company LLC will revise, as appropriate, the W(z) curves to ensure they are representative of the current core conditions should the value of W(z)*[Predicted P(z) / Measured P(z)] become less than 1.04. The revised W(z) curves will be calculated prior to performance of the next required surveillance. Since the W(z) function is set to 1.0 near the top and bottom of the core, this commitment does not apply to the FQ(z) measured in the exclusion zones.

The staff concludes that the licensee's use of the methods documented in WCAP-10216-P-A, and the licensee approach to ensure W(z) curves are representative of core conditions are acceptable.

3.2.1.3 Westinghouse Reload Methodology (WCAP-9272-P-A)

The Westinghouse reload evaluation methodology is documented in WCAP-9272-P-A (Reference 23). This method is based on the concept of a bounding analysis. The method assumes that the validity of the reference analysis is established for the reload core under consideration, on the basis that the key safety parameters for the reload core use values that are conservatively bounded by those in the reference analysis. For each reload core, the values of the key safety parameters are examined to determine whether a transient analysis is required or not. If all key safety parameters are conservatively bounded, the reference safety analysis remains valid for the reload core. If a reload parameter is not bounded, further analysis or evaluation is required for the reload core. For each design-basis event, the method identifies the cycle-dependent safety parameters and their limiting directions. The NRC staff found that the methodology was previously approved by the NRC for Westinghouse plants in performing reload analysis (Reference 23). Since the reload evaluation methodology is sensitive to the set of computer codes methods of analysis being applied, the NRC staff's SE approving the methodology requires that any significant change in methods or codes by Westinghouse must be evaluated for its impact on the reload SE methodology of WCAP-9272. The licensee proposes to use the NRC-approved Westinghouse codes and methods to perform its reload analysis. The NRC staff evaluation (discussed in Section 3.2.1.4) determines that the Westinghouse codes and methods are acceptable for use in licensing applications. Therefore, the NRC staff concludes that the licensee use of the Westinghouse reload methodology is acceptable.

3.2.1.4 Computer Codes and Models Used for non-LOCA Safety Analyses

As indicated in Attachment 1 to Reference 17, the licensee performs reload analyses with the following computer codes and models:

- VIPRE-01 (WCAP-14565-P-A): This code is used to perform core thermal-hydraulic analyses, which determines coolant density, mass velocity, enthalpy, vapor void, static pressure and the DNBR distribution along parallel flow channels within the reactor core under normal operational and transient conditions. As documented in WCAP-14565-P-A (Reference 25), the code has been approved by the NRC for use to support operation of Westinghouse plants. The CPSES, Units 1 and 2 have a Westinghouse designed core and its core analysis (discussed in Section 3.2.2) shows that the core thermal-hydraulic conditions during transients are within the NRC-approved range of the code. In addition, by Reference 17, the licensee demonstrated compliance with the SE restrictions associated with WCAP-14565-P-A (Reference 25). Therefore, the NRC staff concludes that the application of VIPRE-01 for the CPSES, Units 1 and 2 core thermal-hydraulic calculations is acceptable.

- WRB-2 and W-3 Critical Heat Flux (CHF) Correlations: The safety DNBR limits have been imposed to assure that there is at least a 95 percent probability at a 95 percent confidence level that the hot rods in the core do not experience a DNB during a transient. For CPSES, Units 1 and 2, loaded predominately with the 17x17 VANTAGE+ fuel in the reactor core, the licensee used the VIPRE-01 code and the WRB-2 correlation limit of 1.17 for the DNBR analysis. The use of WRB-2 correlation was previously approved by NRC for the 17x17 VANTAGE+ fuel used in the CPSES, Units 1 and 2 (References 20 and 21). For conditions where the WRB-2 correlation is not applicable, the licensee used the W-3 correlation with a limit of 1.3 (1.45 for pressures between 500 psia and 1000 psia). Specifically, the steam line break analysis used the W-3 correlation with a limit of 1.45 for the RCS pressures in the range of 500 to 1000 psia. The rod withdrawal event from subcritical conditions was analyzed using the W-3 correlation with a limit of 1.3 for locations below the bottom non-mixing vane grid. The safety DNBR limit for the W-3 correlation is 1.3 for the RCS pressures above 1000 psia.

The safety limits for both WRB-2 and W-3 CHF correlations with use of VIPRE-01 were previously approved by the NRC for Westinghouse plants (Reference 25). In addition, CPSES, Units 1 and 2 application of the CHF correlations (discussed in Section 3.2.2) is within the applicable ranges. Therefore, the NRC staff concludes that the use of the correlations with the associated safety DNBR limits is acceptable.

- RETRAN-02 (WCAP-14882-P-A): This code simulates a multi-loop system using a model containing a reactor vessel, hot and cold leg piping, steam generators, and pressurizer. The code also includes point kinetics and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for analyses of thermal transients and water level responses for indication and control. As documented in WCAP-14882-P-A, the code has been generically approved by the NRC for Westinghouse to analyze system responses to non-LOCA transients for Westinghouse pressurized-water reactors (Reference 26).

  In this LAR, the licensee proposes to use RETRAN-02 to perform analyses of the following events: (1) feedwater system malfunctions; (2) steam line break; (3) loss of external electrical load/turbine trip; (4) LOOP; (5) loss of normal feedwater; (6) feedwater pipe break; (7) loss of forced reactor coolant flow; (8) locked pump rotor/shaft break; (9) uncontrolled rod cluster control assembly (RCCA) withdrawal at power; (10) inadvertent operation of ECCS; and (11) inadvertent opening of pressurizer safety or relief valve; and (12) steam generator tube rupture. The NRC staff found that: (1) each event proposed to be analyzed for the CPSES, Units 1 and 2 using RETRAN-02 is listed in Table 1 of the NRC staff's SE approving RETRAN-02; (2) the code is reliable in calculating the system responses during transients; and (3) the results of the transient analysis (discussed in Section 3.2.2) meet the acceptance criteria for the analysis of record. The NRC staff, therefore, concludes that the licensee's use of RETRAN-02 in performing non-LOCA analysis for the CPSES, Units 1 and 2 is acceptable.

- LOFTRAN (WCAP-7907-P-A): The LOFTRAN code provides a simulation of the RCS response and calculates system parameters such as core power, RCS flow, RCS primary and secondary pressures and temperatures. The code was previously approved for Westinghouse plants for use in performing the design-basis non-LOCA transients (Reference 27). The licensee proposed to use LOFTRAN to analyze the rod drop event and anticipated transient without scram (ATWS). The NRC staff found that the events proposed to be analyzed using LOFTRAN are listed in the NRC staff's SE approving LOFTRAN, and the results of the analyses (discussed in Section 3.2.2) show that the acceptance criteria for the analyses are met. The NRC staff, therefore, concludes that the licensee's use of LOFTRAN in performing the rod drop event and ATWS analyses for CPSES, Units 1 and 2 is acceptable.

- TWINKLE (WCAP-7979-P-A): This multi-dimensional spatial neutronics code uses an implicit finite-difference method to solve the two group transient neutronics equations in one, two, and three dimensions. This code is documented in WCAP-7979-P-A (Reference 28). The licensee proposed to apply TWINKLE to the CPSES, Units 1 and 2 for analysis of the uncontrolled RCCA withdrawal from a subcritical condition and the RCCA ejection event. The NRC staff found that the TWINKLE code was previously approved by NRC for Westinghouse plants in calculating the kinetics response of a reactor for transients such as the uncontrolled RCCA withdrawal from a subcritical condition and the RCCA ejection event. Since the TWINKLE code is a generic neutron kinetics code approved by the NRC and the licensee applied TWINKLE using the nuclear characteristics specific to the CPSES, Units 1 and 2 fuel, the application of the TWINKLE code for the proposed use is acceptable.

- FACTRAN (WCAP-7908-P-A): As documented in WCAP-7908-P-A (Reference 29), the NRC has approved the code for Westinghouse plants in calculating the transient heat flux at the surface of a rod. FACTRAN is an NRC-approved code for calculating thermal transients in a fuel rod. In addition, in Reference 17, the licensee demonstrated compliance with the SE restrictions (related to initial fuel temperatures, the gap heat transfer coefficient and number of concentric rings used for a fuel rod, etc.) imposed on use of FACTRAN. Therefore, the NRC staff finds that the licensee's application of FACTRAN is acceptable.

- PHOENIX-P and ANC (WCAP-11596-P-A, WCAP-10965-P-A): Both codes address three-dimensional features of the nuclear characteristics of the fuel. PHOENIX-P is used to generate the cycle-specific nuclear cross sections. ANC with the input from PHOENIX-P is used to calculate the nuclear characteristics such as power distributions, control rod worth, and reactivity feedback coefficients. As documented in WCAP-11596-P-A (Reference 30) and WCAP-10965-P-A (Reference 31), the NRC has approved both PHOENIX-P and ANC for Westinghouse plants in calculating the nuclear characteristics. Since the codes were approved by the NRC and the licensee applied the codes using fuel design characteristics specific to the CPSES, Units 1 and 2, the licensee's application of the codes is acceptable.

- Method for the Rod Ejection Analysis (WCAP-7588): As documented in WCAP-7588 Rev. 1-A (Reference 32), the NRC has approved the method which relies on spatial kinetics models for the Westinghouse plants to perform the rod ejection analysis. Since the methods were approved by the NRC and the licensee applied the method using plant conditions specific to the CPSES, Units 1 and 2, the licensee's application of the method is acceptable.

- Method for the Dropped Rod Analysis (WCAP-11394-P-A): WCAP-11394-P-A (Reference 33) documents the methodology for analysis of the dropped rod event. The NRC staff found that the NRC had previously approved the TR for

Westinghouse plants in performing plant transient analysis. Since the TR was previously approved by the NRC, and CPSES, Units 1 and 2 are Westinghouse-designed plants, the licensee's proposed use of the TR is consistent with current licensing basis and is acceptable.

- Method for the Pressurizer Safety Valve Set Pressure Shift (WCAP-12910 Rev. 1-A): WCAP-12910 Rev. 1-A (Reference 34) documents the methodology used in assessing and applying the pressurizer safety valve response to transient analyses. The NRC staff found that the NRC had previously approved the TR for Westinghouse plants in performing plant transient analyses. Since the TR was previously approved by the NRC, and CPSES, Units 1 and 2 are Westinghouse-designed plants, the licensee's proposed use of the TR is consistent with current licensing basis and is acceptable.

In Reference 2, the licensee evaluated its compliance with the conditions specified in each of the SEs approving Westinghouse TRs applied in the LAR. Accordingly, the NRC staff concludes that the licensee adequately demonstrated conformance to the SE conditions. The licensee performed an analysis of the USFAR limiting events with the Westinghouse methods to demonstrate that the use of the Westinghouse methods is acceptable. The demonstration analysis (discussed in Section 3.2.2) shows that the results meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the application of the generically approved Westinghouse methodologies to the CPSES, Units 1 and 2 for non-LOCA analyses is acceptable.

3.2.1.5 BEACON Core Monitoring and Operation Support System (WCAP-12472-P-A)

In addition to the non-LOCA methodology change, the LAR proposes a change that would allow use of the power distribution monitoring system (PDMS) for power distribution measurements as described in WCAP-12472-P-A (Reference 36). The Best Estimate Analyzer for Core Operation Nuclear (BEACON) system was developed by Westinghouse to improve the monitoring support for Westinghouse-designed PWRs. It is a core monitoring and support package which uses Westinghouse standard instrumentation in conjunction with an analytical methodology for on-line generation of three-dimensional power distributions. The system provides core monitoring, core measurement reduction, core analysis, and core predictions. WCAP-12472-P-A was approved by the NRC staff on February 16, 1994.

CPSES, Units 1 and 2 will use the BEACON system to augment the functional capability of the flux mapping system for the purpose of power distribution surveillances. WCAP-12472-P-A discusses an application of BEACON in which the TS and core power distribution limits are changed to take credit for continuous monitoring by plant operators. CPSES, Units 1 and 2 will use a conservative application of BEACON where the core power distribution limits remain unchanged; referred to as the BEACON TS monitor (TSM). CPSES, Units 1 and 2 will use the BEACON PDMS as the primary method for power distribution measurements and the flux mapping system, if required, when thermal power is greater than 25 percent rated thermal power (RTP). At thermal power levels less than or equal to 25 percent RTP, or when PDMS is inoperable, the moveable incore detector system will be used.

The PDMS instrumentation provides the capability to monitor core parameters at more frequent intervals than is currently required by TS. The PDMS combines inputs from currently installed plant instrumentation and design data for each fuel cycle, and does not modify or eliminate existing plant instrumentation. It provides a means to continuously monitor the power distribution limits including limiting peaking factors and quadrant power tilt ratio. The PDMS instrumentation does not change any of the key safety parameter limits or levels of margin as considered in the reference design basis evaluations. These limits are not revised by this license amendment, and can be determined independently of the operability of the PDMS. PDMS itself does not meet any of the 10 CFR 50.36(d)(2)(ii) selection criteria for inclusion into the TS. Therefore, the PDMS does not need a TS requiring its operability.

The NRC staff has reviewed the licensee's proposal to adopt the use of a BEACON system as a TSM and finds the proposal acceptable. The acceptability of the associated TS changes is discussed in Section 3.2.3 of this SE.

3.2.2 Licensing Basis Events

Events Analyzed

The licensee provided plant-specific information (Reference 17) to justify the acceptability of the generically approved Westinghouse methods for CPSES, Units 1 and 2 application. Specifically, the licensee performed analyses of the limiting UFSAR events with the Westinghouse methods to demonstrate that the use of the methods is within the approved ranges and the results of the analysis meet the applicable acceptance criteria. Those events that are non-bounding were evaluated without explicit calculations and are indicated as "evaluated – non-bounding."

The events analyzed include:

- Decrease in Feedwater Temperature
- Increase in Feedwater Flow
- Excessive Increase in Secondary Steam Flow (evaluated – non-bounding)
- Inadvertent Opening of a Steam Generator Relief or Safety Valve (evaluated – non-bounding)
- Steamline break from 0 percent power and 100 percent power
- Loss of External Electrical Load (evaluated – non-bounding)
- Turbine Trip
- Inadvertent Closure of Main Steam Isolation Valves (evaluated – non-bounding)
- Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip (evaluated – non-bounding)
- Loss of AC to the Station Auxiliaries

- Loss of Normal Feedwater
- Feedwater System Pipe Break
- Partial Loss of Forced Reactor Coolant Flow
- Complete Loss of Forced Reactor Coolant Flow
- Locked Rotor/Shaft Break
- Uncontrolled RCCA Withdrawal from a Subcritical Condition
- Uncontrolled RCCA Withdrawal at Power
- RCCA Misalignment
- Chemical and Volume Control System Malfunction (Boron Dilution)
- RCCA Ejection
- Inadvertent Operation of ECCS During Power Operation
- Inadvertent Operation of Pressurizer Safety or Relief Valve
- ATWS

Key Plant Parameters

Key plant parameters assumed in the non-LOCA transient analyses are as follows:

- NSSS full-power (with RCP heat) of 3628 MWt.
- nominal full-power reactor coolant vessel average temperature (T_{avg}) range of 574.2 °F to 589.2 °F, to cover the two units.
- RCS thermal design flow (TDF) of 95,700 gallons per minute per loop (gpm/loop) or 382,800 gpm total.
- maximum SG tube plugging of 10 percent.
- nominal pressurizer pressure of 2,250 psia.
- radial peaking factor of 1.538 for analyses using the revised thermal design procedure (RTDP) and 1.60 for non-RTDP analyses.
- total peaking factor of 2.50.

Most transients analyzed for DNB concerns used the RTDP methodology (Reference 24). With the RTDP method, nominal values were assumed for the initial conditions of power, temperature, pressure, and flow. The corresponding uncertainties were accounted for statistically in determining the DNBR design safety limit. The nominal flow assumed in RTDP transient analyses was the minimum measurement flow of 396,400 gpm.

For transient analyses that were not DNB-limited, or for which RTDP was not used, the initial conditions were obtained by applying the maximum uncertainties to the nominal values in the most conservative direction. In these analyses, the RCS flow was assumed to be equal to the TDF, and the following uncertainties were used:

- ±2 percent NSSS power level uncertainty
- ±4°T_{avg} temperature measurement uncertainty
- ±30 pounds per square inch pressurizer pressure measurement uncertainty

Consistent with the Westinghouse reload evaluation methodology described in Reference 23, the values of kinetics parameters (e.g., control rod worth, reactivity feedback coefficient, shutdown margin) used in the analysis were selected to conservatively bound those expected in subsequent operating cycles. A +3 percent setpoint tolerance was included in the modeling of the main steam safety valves (MSSVs) and pressurizer safety valves (PSVs). The valves setpoints were assumed in the most conservative direction for each case. For example, for the loss of external load event, valves setpoints with inclusion of positive tolerances increase effective MSSV and PSV opening pressures, which would result in a higher increase in the RCS pressure. Therefore, for the overpressure consideration, the analysis assumed that the valves opened at the setpoints with inclusion of positive tolerances. Use of valves setpoints with inclusion of negative tolerances lowers effective valves opening pressures, which would cause an earlier opening of the MSSVs and PSVs, and a lower increase in the RCS pressure. A lower RCS pressure could result in a lower DNBR. Therefore, for the DNBR calculations, the analysis considered negative tolerances for opening of the MSSVs and PSVs.

The NRC staff reviewed the non-LOCA analyses and summarized the computer codes used for each case and the results of the analysis for the most limiting events in Table 3. The NRC staff finds that: (1) the licensee used the approved codes and methodologies to perform transient analyses; (2) the values used for the input parameters are conservative in predicting the worst consequences; (3) the computer codes are reliable in calculating core power, pressure, temperature, RCS flow and pressurizer water volume during the transient; and (4) the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and GDC 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Therefore, the NRC staff concludes that the analyses are acceptable.

TABLE 3 - Limiting Non-LOCA Analysis Results and Computer Codes and Critical Heat Flux Correlations Used in the Analysis

Event Analyzed	Code(s) Used	Staff Approval	Key Parameter(s)	Result vs. (Acceptance Criteria)
Decrease in Feedwater (FW) Temperature	RETRAN	Reference 26	minimum departure from nucleate boiling ratio (MDNBR) (WRB-2)	1.90 (1.61)
Increase in FW flow	RETRAN/ VIPRE	References 25 and 26	MDNBR HFP (WRB-2) MDNBR HZP (W-3)	2.10 (1.61) Non-limiting
Steamline break at 0% power	RETRAN/ VIPRE	References 25 and 26	MDNBR (W-3)	2.86 (1.45)
Steamline break at 100% power	RETRAN/ VIPRE	References 25 and 26	MDNBR (WRB-2)	1.96 (1.61)
Turbine Trip	RETRAN/ VIPRE	References 25 and 26	MDNBR (WRB-2) Peak RCS Pressure Peak MSS Pressure	1.98 (1.61) 2746 (2748) 1298 (1318)
Loss of Normal FW	RETRAN	Reference 26	Maximum Pressurizer mixture vol, ft3	1600 (1800)
FW line break	RETRAN	Reference 26	Minimum margin to Hot Leg Saturation, °F	10 (>0)
Partial Loss of Flow	RETRAN/ VIPRE	References 25 and 26	MDNBR (WRB-2)	2.17 (1.61)
Complete Loss of Flow	RETRAN/ VIPRE	References 25 and 26	MDNBR (WRB-2)	1.90 (1.61)
Locked Rotor	RETRAN/ VIPRE	References 25 and 26	Peak RCS Pressure Peak Clad Temp, °F Maximum Zr-Water Reaction, % Maximum % Rods in DNB, %	2575 (2748) 1724 (2700) 0.22 (16) 0.135 (10)
Uncontrolled RCCA Withdrawal from subcritical	TWINKLE/ FACTRAN/ VIPRE	References 25, 28, and 29	MDNBR below mixing vane grid (W-3) MDNBR above first mixing vane grid (WRB-2) Maximum Fuel Centerline Temp, °F	1.62 (1.3) 2.00 (1.17) 2304 (4800)
Uncontrolled RCCA Withdrawal from power	RETRAN	Reference 26	MDNBR (WRB-2) Peak MSS Pressure, psia	1.69 (1.61) 1276 (1318)
RCCA Misalignment	LOFTRAN/ ANC/VIPRE	References 25 and 27	MDNBR (WRB-2) Peak LHGR, kw/ft Peak Cladding Strain, %	>1.61 (1.61) <22.4 (22.4) <1.0 (1.0)
Boron Dilution		Reference 35	Time from alarm to operator action to prevent complete loss of shutdown margin, minimum	48.2 (15)
RCCA Ejection	TWINKLE/ FACTRAN	References 28 and 29	Maximum Pellet Ave. Enthalpy, cal/g Maximum % fuel melt	161.6 (200) 0.23 (10)
Inadvertent Operation of ECCS	RETRAN	Reference 26	Maximum Pressurizer mixture vol, ft3	1780 (1800)
Inadvertent Opening of Pressurizer Safety or Relief Valve	RETRAN	Reference 26	MDNBR (WRB-2)	1.9 (1.61)
Steam Generator Tube Rupture	RETRAN	References 26, 37, and 38	Margin to overfill, ft3 Dose, rem	23 (0) < Acceptance limit
ATWS	LOFTRAN	Reference 27	Peak RCS pressure, psig	<3200 for 95% of cycle (<3200 for 95% of cycle)

3.2.2.1 ATWS Analysis

An ATWS event is defined as an anticipated operational occurrence (AOO) combined with an assumed failure of the reactor tip system to shutdown the reactor. For Westinghouse plants, the ATWS Rule, 10 CFR 50.62, requires the installation of an ATWS mitigating system actuation circuitry (AMSAC) system to initiate a turbine trip and actuate auxiliary feedwater (AFW) flow independent of the reactor protection system. The licensee has met the ATWS rule by installing an AMSAC at the CPSES, Units 1 and 2. In support of the proposed LAR to convert to Westinghouse non-LOCA methods, the licensee reanalyzed the Loss of Load and Loss of Normal Feedwater ATWS events to confirm that the analytical basis for the ATWS rule continues to be met. The licensee analyzed the ATWS events using the NRC-approved LOFTRAN code.

In accordance with the NRC staff review position on the ATWS analysis for the existing PWRs, such as the CPSES, Units 1 and 2, the NRC staff considered that an acceptable ATWS analysis would show that the unfavorable exposure time (UET), given the cycle design, including the moderator temperature coefficient, would not be greater than 5 percent, or the ATWS pressure limit would be met for at least 95 percent of the fuel cycle. The UET is the time during the cycle when reactivity feedback is insufficient to maintain pressure under 3200 psia for a given reactor state.

The licensee proposes to demonstrate compliance with the ATWS Rule on a cycle-by-cycle basis by applying the Unfavorable Exposure Time (UET) and critical power trajectory (CPT) methodologies. The CPTs are calculated loci of plant conditions (e.g., power vs. inlet temperature) that result in an RCS pressure of 3200 psig given an ATWS event.

The NRC staff found that for CPSES, Units 1 and 2: (1) the ATWS analysis is performed with the NRC-approved LOFTRAN code, (2) the compliance to the ATWS Rule is demonstrate by applying the UET and CPT methodologies previously approved by the NRC staff, (3) and a cycle-specific analysis or confirmation will ensure that UET will be less than 5 percent, or equivalently the ATWS pressure limit of 3200 psig will be met for at least 95 percent of the cycle. Therefore, the NRC staff concludes that the ATWS analysis is acceptable.

3.2.3 Related Technical Specification Changes

3.2.3.1 TS 5.6.5b Core Operating Limits Report (COLR)

TS 5.6.5.b provides a list of TRs documenting the NRC-approved methodologies used to determine the values of cycle-specific parameters included in the COLR. The LAR proposes to add the following non-LOCA related TRs to the reference list:

- WCAP- 11397-P-A, "Revised Thermal Design Procedure," April 1989.

- WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

- WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

- WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," April 1999.

- WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

As discussed in Sections 3.2.1 and 3.2.2 of this SE, the NRC staff finds that the methodologies documented in the referenced TRs are acceptable for use in support of CPSES, units 1 and 2 licensing applications. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

3.2.3.2 TS 3.2 Power Distribution Limits

The LAR proposes to revise TS Section 3.2 based on methodologies used to establish the core operating limits as described in WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification." The proposed change is also based on NUREG-1431 Vol. 1, Rev. 3 (Improved Standard TS (ISTS) for Westinghouse Plants). The differences relative to the ISTS are around CPSES's use of N-16 trip system, retaining the existing licensing basis for CPSES, Units 1 and 2 and implementing the limited application of BEACON PDMS.

As discussed in Sections 3.2.1.1 of this SE, the NRC staff finds the N-16 based trip system used at CPSES, Units 1 and 2 acceptable. In addition, the proposed changes are consistent with previously approved RAOC methodology and ISTS for Westinghouse plants. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

3.2.3.3 BEACON Related Changes (TS 3.1.7, 3.2.1, 3.2.2, 3.2.4, and 3.3.1)

The LAR proposes to revise TS 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor," TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," and TS 5.6.5, "Core Operating Limits Report," consistent with the technical requirements of WCAP-12472-P-A and 10 CFR 50.36. These changes allow use of either the PDMS or the moveable incore detectors to meet the requirements. The NRC staff has reviewed the licensee's proposed TS changes to allow use of a BEACON system as a TSM and finds the proposal acceptable.

4.0 REGULATORY COMMITMENTS

- The licensee will submit to the NRC the data regarding the measurements and results from TS SR 3.2.1.1 following 6 months of Unit 2 Cycle 11 operation.

- Luminant Generation Company LLC will revise, as appropriate, the W(z) curves to ensure they are representative of the current core conditions should the value of W(z)*[Predicted P(z) / Measured P(z)] become less than 1.04. The revised W(z) curves will be calculated prior to performance of the next required surveillance. Since the W(z) function is set to 1.0 near the top and bottom of the core, this commitment does not apply to the FQ(z) measured in the exclusion zones.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to 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 August 14, 2007 (72 FR 45461). 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) 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. Letter logged TXX-07063 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated April 10, 2007, transmitting License Amendment Request (LAR) 07-003 (ADAMS Accession No. ML071070307).

2. Letter logged TXX- 07107 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated July 31, 2007, transmitting supplemental information for LAR 07-003 (ADAMS Accession No. ML072200641).

3. Bajorek, S. M., et. al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary).

4. Dederer, S. I., et. al., 1999, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1 and WCAP-14450, Revision 1 (Non-Proprietary).

5. Nissley, M. E., et. al., 2005, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-Proprietary).

6. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

7. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

8. Thompson, C. D. et. al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Rev. 1 (proprietary), July 1997.

9. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

10. Letter logged TXX-07151 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated November 15, 2007, "Response to RAI Related to Large and Small Break LOCA" (ADAMS Accession No. ML073241095).

11. Letter logged TXX-07168 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated November 19, 2007, "Response to RAI Related to Large and Small Break LOCA" (ADAMS Accession No. ML073330059).

12. Letter dated August 1, 2005, from R. A. Gramm, U. S. NRC to J. A. Gresham, Westinghouse Electric Company, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, 'Post LOCA Long Term Cooling Model' Due to Discovery of Non-conservative Modeling Assumptions During Calculations Audit" (ADAMS Accession No. ML051920310).

13. Letter dated October 3, 2006, from Sean E. Peters, Project Manager, Special Projects Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, NRC to Stacey L. Rosenberg, Chief, Special Projects Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, NRC, "Summary of August 23, 2006, Meeting With The Pressurized Water Reactor Owners Group to Discuss the Status of Program to Establish Consistent Criteria For Post Loss-Of-Coolant (LOCA) Calculations" (ADAMS Accession No. ML062690017).

14. Letter L-05-169, First Energy Nuclear Operating Company to USNRC, "Responses to a RAI (RAI dated September 30, 2005) in support of License Amendment Request Nos. 302 and 173," November 21, 2005 (ADAMS Accession No. ML053290133).

15. Letter L-05-112, First Energy Nuclear Operating Company to USNRC, "Responses to a RAI in Support of License Amendment Request Nos. 302 and 173," July 8, 2005 (ADAMS Accession No. ML051940575).

16. Attachment 2 to Letter logged TXX-07126 from Mike Blevins of Luminant Power to the NRC, dated August 16, 2007, Supplement to LAR 07-003 (ADAMS Accession No. ML072330512).

17. Letter logged TXX-07126 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated August 16, 2007, transmitting supplementary information to LAR 07-003 (ADAMS Accession No. ML072340228).

18. Letter logged TXX-07163 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated November 15, 2007, Transmitting Response To Request For Additional Information Associated with Methodology Used to Establish Core Operating Limits for LAR 07-003 (ADAMS Accession No. ML073241097) (Non-Publicly Available).

19. WCAP-8745-P-A, "Design Bases for Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

20. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10444, VANTAGE+ Fuel Assembly."

21. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-12610, VANTAGE+ Fuel Assembly Reference Core Report," dated July 1, 1991.

22. RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," dated October 1993.

23. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

24. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1987.

25. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurizer Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis," dated October 1999.

26. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

27. WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

28. WCAP-7979-P-A, "TWINKLE - A Multidimensional Neutron Kinetics Computer Code," January 1975.

29. WCAP-7908-A, "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," December 1989.

30. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurizer Water Reactor Cores," June 1988.

31. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

32. WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.

33. WCAP-11394-P-A, “Methodology for the Analysis of the Dropped Rod Event,” January 1990.

34. WCAP-12910 Rev. 1-A, “Pressurized Safety Valve Set Pressure Shift,” May 1993.

35. RXE-94-001-A, “Safety Analysis of the Postulated Inadvertent Boron Dilution Event in Modes 3, 4, 5,” February 1994.

36. WCAP-12472-P-A, "BEACON: Core Monitoring and Support System," August 1994.

37. WCAP-10698, “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” August 1987.

38. WCAP-10698 Supplement 1, “Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident,” March 1986.

39. Letter logged TXX-08024 from Mike Blevins of Luminant Generation Company LLC to the NRC, dated February 11, 2008, Transmitting Response To Request For Additional Information Related To LAR 07-003 Revision To Technical Specifications 3.1, "Reactivity Control Systems," 3.2, "Power Distribution Limits," 3.3, "Instrumentations," and 5.6.5b, "Core Operating Limits Report (COLR)" (ADAMS Accession No. ML080520043).

40. WCAP-10216-P-A, “Relaxation of Constant Axial Offset Control/FQ Surveillance Technical Specification,” February 1994.

41. Letter logged TXX-08032 from F. Madden of Luminant Generation Company LLC to NRC, dated March 6, 2008, Supplement to License Amendment Request (LAR) 07-003, Response to Request for Additional Information Related to License Amendment Request Associated with Methodology Used to Establish Core Operating Limits (ADAMS Accession No. ML080710563).

42. Letter logged TXX-08051 from R. Flores of Luminant Generation Company LLC to NRC, dated March 13, 2008, Supplement to License Amendment Request (LAR) 07-003, Response to Request for Additional Information Related to Review Associated with Large and Small Break LOCA Analyses (ADAMS Accession No. ML080840547).

43. Letter logged TXX-08054 from R. Flores of Luminant Generation Company LLC to NRC, dated March 26, 2008, Supplement to License Amendment Request (LAR) 07-003, Response to Request for Additional Information Related to License Amendment Request Associated with Methodology Used to Establish Core Operating Limits (ADAMS Accession No. not available at the time of issuance).

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