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Ref. # 10CFR50.90
10CFR50.46

November 15, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
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
(TAC Nos. MD6212 and MD6213)

REFERENCE:
1. Letter logged TXX-07063 dated April 10, 2007 submitting License Amendment Request (LAR) 07-003 revision to Technical Specification 3.1, "REACTIVITY CONTROL SYSTEMS," 3.2, "POWER DISTRIBUTION LIMITS," 3.3, "INSTRUMENTATION," and 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)," from Mike Blevins to the NRC.
2. Letter logged TXX-07107 dated July 31, 2007 submitting the Comanche Peak Units 1 and 2 Large and Small Break LOCA Analyses from Mike Blevins to the NRC.

Dear Sir or Madam:

Per Reference 1, Luminant Generation Company, LLC (Luminant Power) submitted proposed changes to the Comanche Peak Steam Electric Station, herein referred to as Comanche Peak Nuclear Power Plant (CPNPP), Unit 1 and Unit 2 Technical Specifications to allow the use of several Nuclear Regulatory Commission (NRC) approved accident analysis methodologies to be used to establish core operating limits. Included in that submittal were different methods for analyzing the small break loss of coolant accident (LOCA) and the large break LOCA. As prescribed in the NRC's Safety Evaluations approving the generic use of these methods, and in compliance with 10CFR50.46, Luminant Power transmitted, per Reference 2, the evaluation models and results developed in accordance with those methodologies for NRC review.

On November 2, 2007, the NRC provided Luminant Power with a request for additional information regarding the large and small break loss-of-coolant accidents for Comanche Peak. The response to these questions is provided in an attachment to this letter.

In accordance with 10CFR50.91(b), Luminant Power is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new licensing basis commitments regarding Comanche Peak Units 1 and 2.

A001
NRR

Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

I state under penalty of perjury that the foregoing is true and correct.

Executed on November 15, 2007.

Sincerely,

Luminant Generation Company LLC

Mike Blevins

By: Rafael Flores

Rafael Flores
Site Vice President

Attachment - Request for Additional Information Related to Review Associated with Large and Small Break LOCA Analyses

c - E. E. Collins, Region IV
B. K. Singal, NRR
Resident Inspectors, Comanche Peak

Alice Rogers
Environmental & Consumer Safety Section
Texas Department of State Health Services
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REQUEST FOR ADDITIONAL INFORMATION RELATED TO REVIEW ASSOCIATED WITH LARGE AND SMALL BREAK LOCA ANALYSES

OVERALL APPLICABILITY OF WESTINGHOUSE (W) LOSS-OF-COOLANT ACCIDENT (LOCA) METHODS TO CPNPP

1. **In order to show that the referenced generically approved large break LOCA (LBLOCA) and small break LOCA (SBLOCA) analysis methodologies apply specifically to the CPSES plants, provide a statement that the existing interface process between the licensee and its vendor (Westinghouse) include mechanisms which assure that the ranges and values of the input parameters for the CPSES LBLOCA and SBLOCA analysis bound the ranges and values of the as-operated plant parameters. Furthermore, if the CPSES plant-specific analyses are based on the model and or analyses of any other plant, then justify that the model or analyses apply to CPSES. (For example, if the other plant has a different vessel internals design then the model wouldn't apply to CPSES.)**

   Response:
   The design and operating limits used at CPNPP were formally transmitted, in accordance with the CPNPP Quality Assurance program, to Westinghouse for use in the development of CPNPP models. Westinghouse processes employed during the development of the CPNPP models include checks to ensure that the transmitted parameters fell within the ranges for which the methodologies were approved. The SBLOCA and LBLOCA models for CPNPP were developed based on plant-specific geometry and design parameters.

2. **Were any modifications made to the emergency core cooling system (ECCS) licensing models subsequent to the latest NRC approval and applied to the CPSES SBLOCA and/or best estimate LBLOCA analyses? Please identify any changes.**

   Response:
   The ECCS licensing models and evaluation results, submitted to the NRC via NRC Reference 1, were performed in accordance with the NRC-approved methodologies identified in NRC Reference 2; no modifications to the evaluation models have since been made.

APPLICABILITY OF W LOCA METHODS TO FUEL DESIGNS

3. **Describe the core fuel configuration for CPSES Units 1 & 2.**

   Response:
   The fuel designs currently used in the CPNPP Unit 1 and Unit 2 reactor cores are all of Westinghouse designs. The more recent assemblies are of the Vantage+ design; a few older assemblies are of the Optimized Fuel Assembly design. The rods are all of the same diameter and the mixing grids are of the same design; however, the Vantage+ design has intermediate flow mixing grids. There are no fuel assemblies from a non-Westinghouse fuel vendor.

4. **Confirm if the licensee plans to apply W LOCA methods for the CPSES cores loaded entirely with W design fuels.**

Response:
The fuel designs used in the CPNPP core configurations are all of standard Westinghouse fuel products. There are no fuel assemblies from a non-Westinghouse fuel vendor.

5. **If the licensee plans to operate CPSES with mixed-core (fuel designs of different vendors), then provide the following additional information:**

a) Describe the core configuration, including types of vendor fuel designs, percent of each fuel types and the number of cycles exposed.

b) Describe in detail the applicability of W LOCA methods on other vendor's fuel.

c) In addition to W LOCA methods, will it be necessary to apply other vendor's methodologies to perform CPSES large or small break LOCA analysis in a mixed-core? If so, describe how the results will be interpreted, particularly, in determining the values of limiting plant parameters, such as PCT.

d) Describe in detail how a mixed-core analysis using W methods (or in combination with other vendor's methods) accurately models the interactions (neutronic, thermal-hydraulic, etc.) between fuel assemblies of different designs.

[Note: If the licensee plans to operate in the future with mixed-core using W methodology, then prior staff approval is required.]

Response:
Not applicable. See the response to Question 4.

BEST ESTIMATE LARGE BREAK LOCA (BE-LBLOCA)

6. **In Table 1 of Attachment 1 to Reference 1, reactor power used for BE-LBLOCA was indicated to be ≤ 3612 ± 0.6% Mwt (megawatts thermal). Clarify, exactly at what core power level the BE-LBLOCA analysis was performed using ASTRUM code.**

Response:
The ASTRUM BELOCA analyses for the Comanche Peak Units are performed at a nominal core power of 3612 Mwt. The core power is a sampled parameter within the approved ASTRUM methodology, and the sampling range applied for CPNPP is ± 0.6% uncertainty around the nominal core power. (Reference WCAP-16009-P-A, Table 1-10)

7. **In page E-5 of NUREG/CR-5249 (CSAU Methodology), it is stated, "These studies consider the core power peaking determined to be the worst in terms of peak clad temperature, the worst single failure, loss of off-site power and maximum initial plant power accounting for uncertainties in the plant instrumentation which measures plant power." In light of this statement, the staff believes that the maximum initial power used for the BE-LBLOCA analysis should be equal to 3612 + 0.6% Mwt. Explain.**

Response:
Consistent with the above, each of the 124 cases executed in the BE-LBLOCA analysis of each Comanche Peak Unit exhibits a core power level somewhere between 3590.3 Mwt and 3633.7 Mwt, as determined via random sampling. Therefore, the statement that "the maximum initial power used for the BE-LBLOCA analysis" among all of the cases performed equals {3612 + 0.6%} Mwt is correct.

8. **Clarify if the W BE-LBLOCA results presented in Table 2 of Attachments 1 & 2 [Ref. 1] are for the case with or without offsite power.**

Response:
The confirmatory studies performed for each Comanche Peak Unit established that Loss-Of-Offsite-Power is the limiting condition. Therefore, per the ASTRUM methodology, all of the cases performed in the W BE-LBLOCA analysis (and presented in each Table 2) are for the case without offsite power.

9. **Figure 19 of Attachments 1 & 2 [Ref. 1] presented BE-LBLOCA analysis axial power shape operating space envelope for CPSES Units 1 & 2. Explain how it was determined that it includes the bounding power shape.**

Response:
In the ASTRUM statistical methodology, power shape parameters PBOT and PMID are randomly sampled within the axial power envelope of Figure 19 for each case that is executed. The PBOT vs. PMID operational space is derived from the physics of the core designs. The axial power distribution for each of the 124 cases is then generated automatically (Reference WCAP-16009-P-A, Table 1-10) in the ASTRUM process.

10. **Given that the major plant parameter assumptions used in the W BE-LBLOCA analysis for Units 1 and 2 were essentially identical, as shown in Table 1 of Attachments 1 & 2 [Ref. 1], list the key differences between the two units that cause the BE-LBLOCA results to be significantly different between the two units.**

Response:
The key difference between the two units that leads to the observed large break LOCA behaviors is the steam generator design: CPNPP Unit 1 is equipped with replacement Westinghouse Delta-76 units, while Unit 2 is equipped with the original Westinghouse Model D5 units. The Delta-76 steam generators have significantly greater tube flow area than the Model D5 units, and correspondingly lower flow resistance, which leads to an improved prediction of core reflood behavior and consequently a lower peak cladding temperature for the Unit 1 limiting case.

11. **For the purpose of comparing the results of W LOCA methodology with that of the current LOCA analysis of record, please provide the following additional information:**

    a) **PCT, Local Maximum Oxidation (LMO) and Core Wide Oxidation (CWO) for Units 1 and 2 that was calculated for the current limiting LBLOCA and SBLOCA using the currently approved LOCA methodology.**

Response:
The requested values are given in Table 1 below. These values are substantially higher than those in the current submittal for the following reasons.

LBLOCA – The previous AOR results are obtained with an Appendix K methodology. The current results are obtained with a best estimate methodology. While there are many differences between these approaches, a simplistic view is that in the former, all parameters are applied simultaneously at extreme (limiting) values, while in the latter parameter values are randomly generated over realistic statistical distributions. Just as an example, the 1.2x multiplier on decay heat mandated in Appendix K is not required in the best estimate calculation; that difference alone would result in significantly lower LBLOCA results.

SBLOCA – While both previous and current SBLOCA results are obtained with Appendix K methodologies, the previous results are higher mostly because the depressurization rate after loop seal clearing is lower in the previous methodology resulting in the accumulator injection occurring later relative to the NOTRUMP calculation.

Table 1 – AOR results calculated with prior methodologies

	LBLOCA			SBLOCA		
	PCT (F)	LMO (%)	CWO (%)	PCT(F)	LMO (%)	CWO (%)
Unit 1	2021	4.1	0.40	2080(*)	1.8	0.21
Unit 2	2057	4.5	0.44	1832	2.3	0.26

(*) Includes a 250^{0}F penalty

SMALL BREAK LOCA (SBLOCA)

12. Given that the input parameters used in the SBLOCA analyses for Units 1 and 2 are same, as shown in Table 1 of Attachment 3 [Ref. 1], list the key differences between the two units that cause the SBLOCA results to be significantly different between the two units.

Response:
The steam generator designs are different. Specifically, Unit 1 has Delta-76 steam generators and Unit 2 has D-5 steam generators. The aspect of this difference that is most relevant to the SBLOCA analysis is that the Delta-76s have more tube area/volume which increases the RCS volume by approximately 10% over the D-5s. Nevertheless, the responses of the two units do not differ qualitatively, i.e., in terms of the phenomena and the sequence in which they occur. The differences that do occur are quantitative, i.e., in timing and magnitude. These differences are not considered significant and can be explained by the steam generator differences described above. The fact that the differences are not that great can be observed by comparing the various parameter histories. See also the answer to question 18 regarding the similarities between the units' SBLOCA responses.

13. **As part of Condition 1 for applicability of W SBLOCA methodology, it was stated in page 15 of Attachment 1 [Ref. 2], "To assure the validity of this application, the bubble diameter should be on the order of 10-1 to 2 cm." Clarify what is meant by 10-1 to 2 cm.**

Response:
10-1 should read 10^{-1} (i.e., 0.1 cm) to 2 cm or one tenth of a centimeter to two centimeters.

14. **As part of the justification for Condition 3 for W SBLOCA, it was stated in page 16 of Attachment 1 [Ref. 2], "The LOCTA code has always accounted for axial conduction as is clearly stated in WCAP 14710, which supplements the original NOTRUMP documentation." The staff noted that the letter "A" did not appear next to the WCAP to signify NRC approval of the report. Confirm if WCAP 14710, and the addition of clad axial heat conduction was approved by NRC.**

Response:
WCAP-14710-P-A, which allows for explicit modeling of annular type fuel pellets is approved and should have been marked as an approved ("-A") report. In addition, note that in the NOTRUMP Evaluation Model (NOTRUMP-EM), the clad axial heat conduction is addressed by the LOCTA code. The use of the LOCTA code in the approved NOTRUMP-EM is described in WCAP-10054-P-A.

15. **In page 2 of Attachment 1 to TXX-07063, it was proposed to revise CPNPP TS 5.6.5b, Core Operating Limits Report (COLR), to add nine NRC approved WCAPs. The list, however, did not include WCAP 14710. Explain.**

Response:
T.S. 5.6.5.b indicates that "the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC... ". The method used to analyze the SBLOCA transient (and therefore determine the core operating limits) is the approved NOTRUMP Evaluation Model (NOTRUMP-EM), which is described in WCAP-10079-P-A, WCAP-10054-P-A and WCAP-10054-P-A, Addendum 2, Revision 1. WCAP-14710-P-A supplements those documents but is not typically included in the COLR list of references. The reasoning is that, as described in question 14, the use of the LOCTA code in the approved NOTRUMP-EM is described in WCAP-10054-P-A. WCAP-14710-P-A merely adds features to LOCTA and is not considered inherent to the methodology so the approval of WCAP-14710 did not result in a change (but rather a supplement) to the SBLOCA methodology used to determine the core operating limits and therefore would not be included in the COLR list of references.

16. **The last sentence of page 2.7-52 of Attachment 2 [Ref. 2] stated, "Most recently, the NRC generically approved updates to the NOTRUMP-EM to include the ability to model annular fuel pellets (WCAP-14710) in the fuel rod heatup calculations." Clarify if CPSES cores include annular fuel pellets.**

Response:
The current (and currently planned) CPNPP core configurations have annular fuel pellets only in the axial blankets (top and bottom 6 inches). These annular fuel pellets

are enriched to a lower value than the central regions of the core to improve neutron economy, and have a negligible impact on SBLOCA results.

17. Were time step studies performed on the limiting small break? Please explain and provide the results of the time step study.

Response:
As explained starting on page 2-16 of WCAP-10079-P-A, NOTRUMP utilizes an implicit solution scheme for the conservation equations. A fully-implicit numerical solution scheme theoretically is not time-step dependent. However, NOTRUMP contains elements such as constitutive equations, boundary conditions and others which are not discretized in a fully implicit manner. As a result, NOTRUMP does require a maximum time step size, such that calculations with values lower than this maximum will yield essentially the same result. These maximum time steps have been established generically as a function of break size and are part of the guidelines for applying the code. Therefore, the time step concern has been addressed because those guidelines were followed, even though time step studies were not performed specifically for this application.

18. The core mixture level predicted by the NOTRUMP-EM displays erratic behavior for all the SBLOCA break sizes, particularly, more so for the limiting break size of 4-inch for Unit 2 (Fig. 3B). Please provide the following additional information:

a) Provide a physical rationale for this behavior.

Response:
Large oscillations in the mixture level occur around 200 seconds (the loop seal clearing time) and are explained in item (b) below. Other oscillations throughout the transient are, for the most part, less than 6 inches in amplitude and reasonable given that the system is undergoing a severe transient. The overall behavior of the mixture level for the 4 inch transient for Unit 2 is explained as follows: The mixture level decreases until around 1000 seconds. Around that time the RCS mass starts to increase due to the break flow being overtaken by the ECCS flow. There are large oscillations in this time window (around 200 seconds) but that is explained in item (b) below. The rate of rise in the mixture level is a bit more rapid around 1000 and 2500 seconds because there is substantial accumulator injection at those times. (Smaller step-like rises in mixture level are co-incident with spurts of accumulator inflow of lesser magnitude.)

b) The core mixture level drops to a minimum level of about 6 ft below the top of core after about 200 seconds from initiation of the transient for a relatively short duration. The level recovers, and then uncovers the core again for a prolonged period of about 1000 seconds before finally quenching the core. Is there a physical rationale behind this prediction? If so, explain. The response should also include why the PCT did not occur during the first core uncovery.

Response:
The first core uncovery, after about 200 seconds, is due to the core level depression associated with the loop seals being plugged. When the broken loop seal clears, the pressure differential between the hot and cold legs drops significantly and abruptly. As

a result, the mixture level in the core increases in order to balance the new loop pressure differential. The loop seal uncovery is an abrupt phenomenon. When it occurs the level is pushed up abruptly. As a result, the momentum associated with the large core fluid mass moving up towards its new equilibrium takes the level beyond the equilibrium point. This overshoot causes the oscillations that are seen around that time. These oscillations dampen out in about 3 minutes. The second core uncovery leads to the PCT and is associated with the boil off. The level depression associated with the broken loop seal clearing typically does not result in a PCT because it is relatively short lived and self-limiting: as the level depression increases, the driving force to clear the loop seal increases and when the broken loop seal finally clears the level has to rise in order that the core delta p match the "new post-clearing" loop delta p. Note that in Unit 2 the level depression associated with the loop seal does not extend below the top of the core, but is present nevertheless. This response was referred to earlier in the answer to Question 12 as a quantitative difference but not a qualitative one. Qualitatively, the transients are essentially identical in both units.

POST-LOCA SUBCRITICALITY CALCULATIONS

19. Section 2.7.3.3 [Ref. 2] provides post-LOCA subcriticality analysis. Describe the specific methodology and computer codes used for this calculation, and whether the methodology and the codes were approved by the NRC.

Response:

The methodology for performing the sump boron concentration calculations is consistent with the approach discussed in Westinghouse Technical Bulletin NSID-TB-86-08 (Reference 3). This methodology supports the licensed Westinghouse Emergency Core Cooling System (ECCS) Methodology reported in WCAP-8339 and WCAP-8471-P-A (References 1 and 2). The results summarized in Section 2.7.3.3 are based on simple hand-calculations and no specific computer codes are used.

The Westinghouse position for satisfying the long-term core cooling criterion of 10 CFR 50.46 is defined in References 1 and 2. Per Reference 1, the continuous supply of borated water in the long-term post-LOCA is adequate to remove decay heat and to ensure the core remains subcritical, assuming that the control rods remain fully withdrawn. Per Reference 2 (p. 3-93), the long-term cooling phase of the LOCA transient is defined as beginning at the time of switchover to sump recirculation.

The post-LOCA sump boron concentration curve as a function of the initial RCS peak xenon boron concentration, as shown in Figure 2.7.3.3-1, is verified by the cognizant core design group for each reload cycle to ensure that core subcriticality in the long term post-LOCA.

Consistent with Reference 3, the calculation considers any source of water that may eventually reside in the containment sump. Boron concentrations are conservatively minimized for all of the volumes considered in this calculation. Mass inventories for these volumes are conservatively minimized for boration sources and conservatively maximized for dilution sources.

Response References:

1. WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model – Summary", F. M. Bordelon et al., June 1974.

2. WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information", F. M. Bordelon et al., April 1975.
3. Thompson, C. M., "Post-LOCA Long-Term Cooling: Boron Requirements", Westinghouse Technical Bulletin NSID-TB-86-08, October 31, 1986.

POST-LOCA LONG-TERM COOLING CALCULATIONS

20. Section 2.7.3.4 [Ref. 2] provides post-LOCA long-term cooling analysis. In page 2.7-59 [Ref. 2] it is stated, "The boric acid precipitation calculation... meets NRC guidance as presented in Reference 3 and is consistent with the interim methodology reported in Reference 4." The staff understands that as part of NRC guidance cited above, Westinghouse-type plants should use the Beaver Valley/Ginna extended power uprate approach. Describe which approach was used for CPSES.

Response:
The approach used for the CPNPP post-LOCA long-term cooling analysis is consistent with the Beaver Valley EPU approach (Reference 1). The most significant assumptions related to the Beaver Valley EPU approach, and used for CPNPP, are provided below:

- Appendix K decay heat
- No credit for mixing in regions outside the mixing volume
- Worst case assumptions for sump boron/water sources
- No credit for entrainment around the loop
- No credit for SI subcooling
- No credit for boron presence in steam
- The effect of core voiding on the liquid mixing volume is calculated using the Modified Yeh Correlation described in Reference 2
- The liquid mixing volume used in the calculation includes 50 percent of the lower plenum volume.
- The boric acid concentration limit is the experimentally determined boric acid solubility limit as reported in Reference 3

Note that the assumptions used in the Beaver Valley EPU approach are discussed in more detail in the response to RAI #F.1 in Reference 4.

Response References:
1. Letter L-05-169, FirstEnergy Nuclear Operating Company to USNRC, "Responses to a Request for Additional Information (RAI dated September 30, 2005) in Support of License Amendment Request Nos. 302 and 173", November 21, 2005.
2. H. C. Yeh, "Modification of Void Fraction Calculation," Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, June 6, 1988.
3. P. Cohen, Water Coolant Technology of Power Reactors, Chapter 6, "Chemical Shim Control and pH Effect," ANS-USEC Monograph, 1980 (Originally published in 1969).
4. Letter L-05-112, FirstEnergy Nuclear Operating Company to USNRC, "Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173", July 08, 2005.

21. In order for the staff to perform an audit calculation to verify CPSES long-term cooling capabilities (boron precipitation), please provide the following information for the CPSES:

a) Volume of the lower plenum, core and upper plenum below the bottom elevation of the hot leg, each identified separately.

Response:

Volume of LP (bottom of vessel to bottom of active fuel)	914.8	ft^3
Volume of Core (active fuel region)	649.6	ft^3
Volume of Upper Plenum (above active fuel and below bottom elevation of HL)	384.6	ft^3

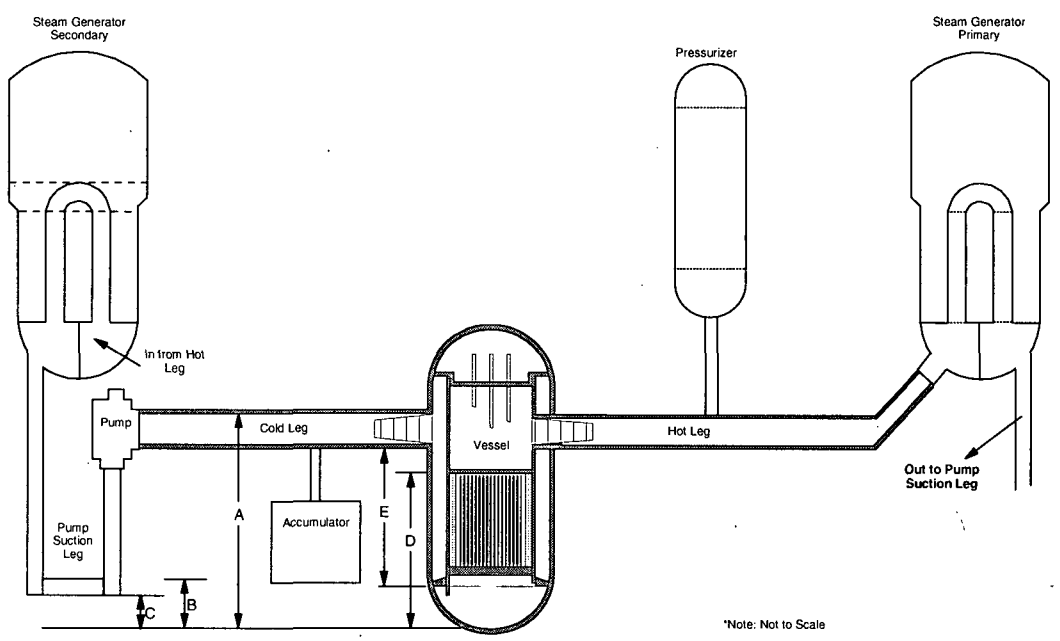
b) Elevations of the top of the RCP discharge leg, the bottom of the suction leg, and top of the core.

Response:

A	Elevation of top of RCP discharge leg	28.49**	ft
B	Elevation of the top of the suction leg	18.32**	ft
C	Elevation of the bottom of the suction leg	15.74**	ft
D	Elevation of the top of the core	22.08**	ft

** Referenced from inside bottom of reactor vessel

Please refer to the following figure for the definitions of A-D in the above table:



c) Capacity of the condensate storage tank.

Response:
The capacity of the condensate storage tank is 249,100 gallons per Technical Specification 3.7.6. A safety-related back-up supply of auxiliary feedwater (AFW) is provided by manual switchover of AFW pump suctions to the Station Service Water System (SSWS), which essentially provides an unlimited back-up supply of AFW. (FSAR Section 9.2.1)

d) Height of the downcomer below the bottom elevation of the cold leg to the inner vessel entry elevation to the lower plenum.

Response:
The height of the downcomer below the bottom elevation of the cold leg to the inner vessel entry elevation to the lower plenum is 19.98 ft.

22. In the long term cooling analysis for boric acid precipitation, did the calculations include the density effect of the boric acid in the mixing volume? This will delay the increase in mixing volume vs. time in the vessel and increase the boric acid concentrations during the first two hours following the LOCA.

Response:
The increased density of the boric acid solution was considered in calculating the mass of the liquid in the mixing volume. The increased density of the boric acid solution on core region mixing volume was not modeled as explained below.

Since the density of a boric acid/water solution increases with percentage of boric acid, there would be a liquid density difference between the core and downcomer. This density difference would be small early in the transient and would increase as the boric acid concentration increases in the core. There may also be some small offsetting density differences due to core/downcomer temperature differences. The net effect of a higher core liquid density as compared to the downcomer would be a lower flooding rate later in the boric acid concentration transient, well after core quench has occurred. The effect of downcomer-core density differences on the core and upper plenum liquid volume can be estimated using the boric acid concentration versus specific gravity plot shown in Figure 1. A 10 wt% boric acid solution has a specific gravity approximately 3% greater than the 1 wt% boric acid solution that would be in the downcomer. Based on a 60% core void fraction and a 30 wt% boric acid solution in the core, the core region liquid volume would be approximately 5.4% lower than otherwise predicted if this density difference is not recognized. This difference in mixing volume is small in comparison to the conservative mixing volume margin used in the analysis.

Density differences (due to temperature or boric acid concentration) would have a beneficial effect on mixing. An increase in the core fluid density due to concentration of debris and chemicals will result in unstable configuration leading to natural convection flow patterns between the core and the less dense lower plenum, which will limit the density buildup. This was demonstrated in the BACCHUS test results previously provided to the NRC in Reference 2.

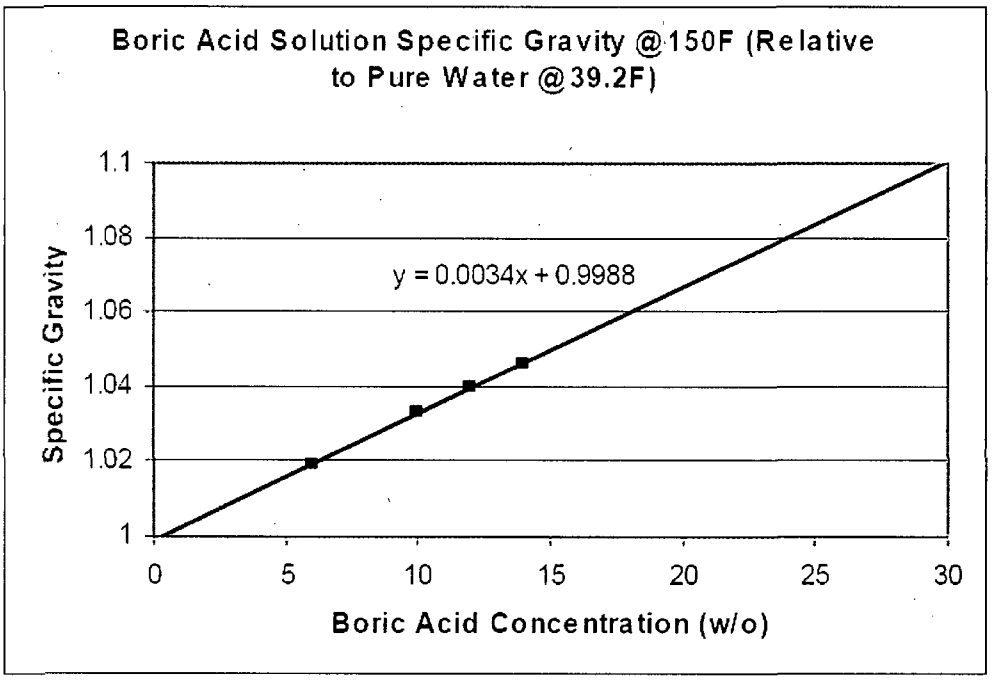


Figure 1
Specific Gravity of Boric Acid – Water Solutions [1]

Response References:
1. "Boric Acid Application Guidelines for Intergranular Corrosion Inhibition", EPRI, Palo Alto, CA: 1987, NP-5558, page 2-27.
2. W3F1-2005-0007, "Supplement to Amendment Request NPF 38 249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3," February 05, 2005.

23. Even though the reactor coolant system (RCS) pressure may remain above 120 pounds-force per square inch absolute as late as 24 hours, is there sufficient condensate for the steam generators to remove heat for this length of time? Please explain.

Response:
Only for the very small breaks is the RCS expected to remain elevated above 120 psia. However, if an extended pressure hang-up were to occur; a safety-related back-up supply of auxiliary feedwater (AFW) is provided by manual switchover of AFW pump suctions to the Station Service Water System (SSWS), which essentially provides an unlimited back-up supply of AFW. (FSAR Section 9.2.1) Therefore, even if the RCS were to remain at an elevated pressure for an extended period of time, there is essentially an unlimited supply of feedwater for the steam generators to remove heat for this period of time. The system will be in natural circulation, and boric acid precipitation is not a concern because the core region is not stagnant. Eventually, subcooled core conditions will be reached, the system will be put into RHR or continued natural circulation and sump recirculation will keep the boric acid from accumulating in the core.

24. Have the LOCA emergency operating procedures been modified to instruct the operators to initiate a cooldown no later than one hour to assure a successful post-LOCA long term cooling? Please note that Fig 2.7.3.4-2 does not provide a maximum time post-LOCA to initiate a cooldown. Please explain.

Response:
The "Small-Break LOCAs" discussion in CPNPP licensing report (Section 2.7.3.4) concludes that the CPNPP post-LOCA long-term cooling analysis makes no assumptions regarding the time for the operators to initiate a cooldown. The discussion provided in the CPNPP licensing report on this scenario is repeated below. It is also important to note that CPNPP has HHSI flow available to provide sufficient dilution flow at these elevated pressures, as shown in Figure 2.7.3.4-4.

"For small breaks (approximately 0.005 - 0.1 ft^2), emergency procedures instruct operators to take action to depressurize and cool down the RCS. Although this depressurization and cooldown process typically begins within one hour after the event, the long-term cooling analysis makes no specific assumptions regarding time to depressurize. Depressurization to 120 psia (the threshold for boric acid precipitation concerns) may occur before or after hot leg switchover time. In either case, the boric acid buildup at hot leg switchover time is conservatively represented by that calculated for the 120 psia RCS backpressure scenario since this calculation takes no credit for SI subcooling, nor any beneficial effects of the operator action (such as reduced net core boiloff due to condensation in the steam generators). If 120 psia is reached before hot leg switchover time, the core dilution flow after hot leg switchover, which is confirmed as adequate for 120 psia backpressure, provides effective core dilution. If at hot leg switchover time, the 120 psia has not been reached, boric acid precipitation does not occur so long as the RCS remains above this pressure since water and boric acid are miscible at the saturation temperature for these pressures. Even if the RCS pressure is above 120 psia at 24 hours after the LOCA with no core dilution flow, the total boric acid in the core is well below the saturation capacity at the corresponding saturation temperature. Furthermore, if after 24 hours with no dilution flow, the RCS is at saturation and depressurized at the maximum cooldown rate, the core is diluted prior to reaching the boric acid precipitation point. If subcooled core conditions are reached either before or after hot leg switchover, boric acid precipitation is not a concern since there is no net boiling in the core. If subcooled core entry conditions are not reached, the operators continue to depressurize the RCS under controlled conditions. Sump recirculation continues, decay heat in the core decreases, and core dilution flow prevents the buildup of boric acid. Eventually, subcooled core conditions are reached, the system is put into RHR or it remains in indefinite recirculation cooling."

25. How much debris from all sources was included in the mixing volume to determine the precipitation timing? Please explain how the impact of debris was included in the assessment.

Response:
The effects of sump debris were not explicitly analyzed in the CPNPP post-LOCA long-term cooling analysis. The conclusions drawn in Section 2.6 of Reference 1 related to the effects of sump debris on boric acid precipitation analyses are assumed to apply to CPNPP. The Reference 1 discussion on post-LOCA long-term cooling concludes that sump debris and related chemical effects do not create a boric acid precipitation concern and that the introduction of debris to the RCS does not significantly affect the

licensing basis boric acid precipitation calculations. Further discussion on the effects of sump debris on long-term cooling analyses is provided in Reference 2 (Response to RAI #9). The Reference 2 discussion reports that the cold leg break scenario of concern for post-LOCA long-term cooling results in minimal debris accumulation in the core. Furthermore, the boric acid precipitation control measures that promote core dilution following a LOCA, will flush the core of concentrated chemicals and suspended debris out of the core region and out the break.

Response References:

1. WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate and Chemical Debris in the Recirculating Fluid", May 2007.
2. Letter OG-07-477, PWROG to USNRC, Pressurized Water Reactors Owners Group Responses to the NRC Request for Additional Information (RAI) on WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate and Chemical Debris in the Recirculating Fluid" (PA-SEE-0312), October 31, 2007.

26. Placing the break on the top of the discharge leg will delay the growth of the mixing volume through the core and upper plenum and increase the concentrations early in the event. Please discuss the effect of breaks located on the top of the discharge leg in determining precipitation timing. Please discuss the growth of the mixing volume vs. time for this scenario (i.e. the two-phase mixture volume in half of the lower plenum and core). At what time in the event does the mixture volume grow sufficiently to reach the upper plenum region?

Response:

NOTE: The response to question 26 contains proprietary information and will be transmitted under separate cover letter.

27. How long does it take to initiate RHR conditions when hot fluid is trapped in the pressurizer for the smallest break that refills and repressurizes the RCS early in the event? How does the operator reduce RCS pressure (what equipment is used and what is the timing for its use) with liquid trapped in the pressurizer that is much hotter than the RCS loop fluid temperature? Is there sufficient condensate storage tank supply to achieve entry conditions into RHR for very small breaks since simply throttling high-pressure safety injection (HPSI) pumps will not reduce RCS pressure to the loop saturation temperature? What guidance is given to the operators to deal with this particular scenario? How are HPSI pumps throttled to achieve RCS RHR pressure conditions once temperature has been met?

Response:

The concern seems to be that the RCS might end up at a pressure above the RHR shutoff heat, i.e. unable to enter shutdown cooling while at the same time it would be without secondary cooling due to CST depletion. With respect to that concern, note that the CPNPP Charging and High Head Safety Injection (HHSI) pumps both are capable of operating in recirculation mode. Both types of pumps, when in recirculation mode, take suction from the RHR pumps, which take suction from the containment. (HHSI can also perform hot leg recirculation. Charging pumps cannot but their capacity is small relative to HHSI at the pressures of interest) Therefore in this mode of operation (hot or cold leg recirculation from the sump) the RHR heat exchangers will cool down the RCS. For all practical purposes, the plant can operate

indefinitely in this mode so that SG cooling is not required to bring the plant conditions down to normal shutdown cooling mode.

It is possible to supplement this RHR cooling in recirculation mode with SG cooling and the CPNPP CST capacity is given in Tech Spec 3.7.6 (Reference 1). That capacity is based on the requirement that CST inventory be capable of meeting the most limiting (in practice item b) of the following: (a) permit the plant to be maintained at zero load hot standby conditions for a minimum of 2 hrs followed by a 5 hrs cooldown, in the event of a Main Steam or Main Feedwater break and (b) permit operation at zero load hot standby for four hours followed by a 5 hrs cooldown period with either only onsite power or only offsite power available with an assumed single failure. Thus, item (b) implies the CPNPP CST has the capacity to remove at least 9 hours of decay heat plus the sensible heat associated with bringing the RCS to cold shutdown. That is more than sufficient in combination with the RHR heat exchangers in recirculation mode, which can do the job alone anyway.

For a SBLOCA, the emergency operating procedures used in response to a loss of primary reactor coolant would direct the operators to cooldown and depressurize the RCS. The switchover of the ECCS to the cold leg recirculation operating mode could occur as early at 1600 seconds if containment sprays are not shut off early.

References:
1. Comanche Peak Nuclear Power Plant (CPNPP) Technical Specifications

28. Does the total loop pressure drop identified in the document pertain to the equivalent of all loops? If so, please provide the total loop pressure drop for a single loop only.

Response:
The total loop pressure drops identified in the licensing report of 1.3E-08 ft/gpm^2 (without RCP locked rotor) and 7.1E-08 ft/gpm^2 (with RCP locked rotor) represent a single loop.

29. How late in the event does the entrainment of the hot side injection no longer become significant? What model is used to determine the potential for entrainment of hot side injection by the core generated steaming rate? Please explain.

Response:
Entrainment calculations were performed in the CPNPP post-LOCA long-term cooling analysis and are discussed in the CPNPP licensing report in the fourth paragraph of the "Results" section, which is summarized below:

"Calculations were performed to support an early switchover to hot leg or simultaneous injection. Two aspects of early switchover were considered – the hot leg entrainment threshold and core cooling... Entrainment threshold calculations similar to those reported in Reference 5 demonstrated that significant hot leg entrainment would not occur after 75 minutes."

These entrainment calculations were performed using the same approach as used in for the Beaver Valley EPU (Reference 1, RAI #F.7). Specifically, the liquid entrainment

threshold in the hot leg can be established from applying the Ishii-Grolmes (Reference 2) or Wallis-Steen (Reference 3) liquid entrainment onset criteria. These entrainment correlations are valid for flow conditions where the liquid phase does not take up a significant volume of the pipe (such as in the hot legs in post-LOCA) and viscous effects in the liquid are not dominant, that is, that the liquid phase is in the turbulent regime. Note that the correlations have very similar form; however, the Ishii-Grolmes entrainment onset criterion uses liquid phase viscosity whereas Wallis-Steen uses gas phase viscosity.

Response References:

1. Letter L-05-112, FirstEnergy Nuclear Operating Company to USNRC, "Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173", July 08, 2005.
2. Ishii, M.; Grolmes, M. A., Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow, AIChE Journal, Vol. 21, No. 2, pp. 308-319, 1975.
3. Wallis, G. B., One-Dimensional Two-Phase Flow, pp. 390-393, 1969.

NRC References

1. **Letter from F. W. Madden to NRC, TXX-07107, "Submittal of the CPNPP Units 1 and 2 Large and Small break LOCA Analyses," July 31, 2007.**

2. **Letter from R. Flores to NRC, TXX-07126, "CPNPP, SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) 07-003 REVISION TO TECHNICAL SPECIFICATION 3.1, "REACTIVITY CONTROL SYSTEMS," 3.2, "POWER DISTRIBUTION LIMITS," 3.3, "INSTRUMENTATION," AND 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)," August 16, 2007.**