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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) 07-004
REVISION TO THE OPERATING LICENSE AND TECHNICAL SPECIFICATION 1.0, "USE AND APPLICATION" TO REVISE RATED THERMAL POWER FROM 3458 MWT TO 3612 MWT. (TAC NOS. MD6615 AND MD6616)

REFERENCE:
1. Letter logged TXX-07106 dated August 28, 2007 from Mike Blevins to the NRC submitting License Amendment Request (LAR) 07-004, proposing revisions to the Operating Licenses and to Technical Specifications 1.0, "USE AND APPLICATION" to revise rated thermal power from 3458 MWT to 3612 MWT
2. Letter Logged TXX-08008 dated January 10, 2008 from Mike Blevins to the NRC submitting a supplement to License Amendment Request (LAR) 07-004

Dear Sir or Madam:

Per Reference 1, Luminant Generation Company LLC (Luminant Power) requested Technical Specification (TS) changes to the Comanche Peak Steam Electric Station, herein referred to as Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 Operating Licenses and to Technical Specifications 1.0, "USE AND APPLICATION" to revise rated thermal power from 3458 MWT to 3612 MWT. Luminant power supplemented that request by responding to NRC Requests for Additional Information (RAI) per Reference 2.

On January 7, 2008, the NRC provided Luminant Power with additional RAIs from the following branches regarding the proposed changes to rated thermal power.

Accident Dose Branch
Containment and Ventilation Branch
Steam Generator Integrity and Chemical Engineering Branch
Balance of Plant Branch

The responses to these questions are provided in the attachment to this letter.

A001
NRR

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

This communication contains no new license basis commitments regarding CPNPP Units 1 and 2.

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 January 31, 2008.

Sincerely,

Luminant Generation Company LLC

Mike Blevins

By: Fred W. Madden

Fred W. Madden
Director, Oversight & Regulatory Affairs

Attachment - Response to Requests for Additional Information

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
1100 West 49th Street
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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

COMANCHE PEAK STRETCH POWER UPRATE

Accident Dose Branch

NRC Question 1.

In WCAP-16840-NP, "Comanche Peak Nuclear Power Plant, Units 1 and 2, Stretch Power Uprate Licensing Report" (SPULR), it is stated that an updated reactor coolant system (RCS) mass is used for the following radiological consequences analyses: Main Steam Line Failures Outside Containment (Section 2.9.2), Reactor Coolant Pump Locked-Rotor Accident (Section 2.9.3), Control Rod Ejection Accident (LR Section 2.9.4), Failure of Small Lines Carrying Primary Coolant Outside Containment (Section 2.9.5), and Steam Generator Tube Rupture (Section 2.9.6). What is the updated RCS mass value?

CPNPP Response:

Sections 2.9.2, 2.9.3, 2.9.4, 2.9.5, and 2.9.6 used the minimum RCS mass of 2.0E8 grams (4.4E5 lbm). The minimum mass results in conservatively high concentrations for events where activity enters the RCS from the fuel. This includes, Reactor Coolant Pump Locked-Rotor Accident (Section 2.9.3), Control Rod Ejection Accident (Section 2.9.4), and the concurrent iodine spike scenario for Main Steam Line Failures Outside Containment (Section 2.9.2), Failure of Small Lines Carrying Primary Coolant Outside Containment (Section 2.9.5), and Steam Generator Tube Rupture (Section 2.9.6). The mass (minimum versus maximum) used to calculate the initial and pre-accident iodine spike RCS activity is unimportant since the concentration of activity in the RCS will be the same whether the minimum or maximum mass is used. The equilibrium and concurrent iodine spike iodine appearance rates were calculated using a conservative maximum RCS mass of 2.6E8 grams (5.7E5 lbm). This use of minimum/maximum RCS mass is consistent with the current licensing basis analyses, see the Reference below.

Reference:
TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 - Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

NRC Question 2.

Considering the updated RCS mass, what sump liquid volume value is used as an assumption for the emergency core cooling system leakage pathway analysis in the loss-of-coolant accident (LOCA) radiological consequences analysis?

CPNPP Response:

The minimum sump mixing volume used in the LOCA radiological consequences analysis (Section 2.9.7) is 1.534E9 cm^3 (4.053E5 gallons).

NRC Question 3.

In SPULR Section 2.9.4, "Radiological Consequences of a Control Rod Ejection Accident," it is stated that the steam releases for the Control Rod Ejection Accident radiological consequences analysis are recalculated and described in Section 2.9.10. However upon review of Section 2.9.10, the steam releases for this analysis was not discussed. Please provide this information.

CPNPP Response:

The steam releases calculated in Section 2.9.10 for the locked rotor analysis (Section 2.9.3) were used in the Rod Ejection analysis (Section 2.9.4) since the cooldown for each analysis is the same in that the RCS is intact and all of the steam generators are available for plant cooldown. The use of the locked rotor steam releases in the rod ejection analysis is consistent with the current licensing basis analysis, see the Reference below.

Reference:
TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 - Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

NRC Question 4.

In SPULR Section 2.9.7, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident," it is stated that an updated modeling was used to calculate the whole body dose to the Control Room operators from external sources. Please provide the calculation and results of this analysis and identify what is updated.

CPNPP Response:

The calculation of the control room whole body dose due to external factors was calculated in four parts which includes the dose due to the cloud of activity immediately outside of the control room, the dose due to streaming of the external cloud through penetrations in the control room, the dose due to activity collected on the control room intake and recirculation filters, and the dose due to activity confined to the containment building. The control room whole body dose was calculated for releases from containment leakage, emergency safety features leakage, and the containment pressure relief line prior to its isolation. The total whole body dose due to external sources is then added to the whole body dose due to activity inside the control room to determine the total whole body dose to operators in the control room. (Note: actual calculation can be made available for staff review upon request)

External Cloud

The whole body dose due to the cloud of activity outside the control room was determined by calculating the dose outside of the control room and applying an attenuation factor to model the dose reduction by the control room walls. The attenuation factors for each of the nuclides were calculated using discrete ordinates techniques for a location adjacent to the minimum thickness concrete envelope (1′ 9″) and represent a set of conservative values for the remainder of the control room interior. These attenuation factors apply in the absence of any consideration of radiation streaming through penetrations in the control room concrete envelope. Instead of using the nuclide specific attenuation factors calculated, the most limiting value was applied to all the nuclides. The most limiting value was 1.7E-3 for Kr-88 which was rounded to 0.002 for use in the analysis.

Streaming of External Cloud

The whole body control room dose due to streaming of radiation from the external cloud was calculated by applying a conservative correction factor to the whole body dose calculated for the external cloud of activity. A series of historical studies conducted by Comanche Peak circa 1987 evaluated the effects of radiation streaming through various penetrations in the control room concrete envelope. The study concluded that the streaming pathways created localized dose increases that could be bounded by an evaluation of doorway E40A in the west wall of the building. Door E40A is a 3'- 4" by 7'- 2" steel structure that is 4.5" thick. A conservative assessment of the streaming through this doorway was completed using two-dimensional discrete ordinates techniques and a multiplicative factor that can be applied to the dose results based on the use of the attenuation factor applied to the external cloud of activity was developed. Based on this factor, the final dose, including a conservative estimate of the effects of radiation streaming is given as follows.

$$D_{Total} = 4.49 * D_0$$

Where D_{Total} and D_0 represent the 30 day dose with and without the effects of radiation streaming and the factor of 4.49 is the increase factor due to streaming through the E40A doorway. The dose from the external cloud due to streaming is conservatively estimated as being equal to the external cloud dose times 3.49. The conservative nature of the streaming adjustment factor is due to the choice of dose location within the control room. The streaming effect was evaluated directly in line with the center of the door at a distance three feet from the interior wall of the control room. The shielding adjustment reduces significantly for locations further from the interior surface of the wall as well as laterally away from the center of the doorway. An additional conservatism in the streaming factor is due to the use of the most severe energy spectrum for the gamma ray emissions for the individual isotopes of concern (in this case, Kr-88).

Activity on Filters

The RADTRAD code was used to determine the build up of iodine activity on the control room filters for the containment and engineered safety feature (ESF) leakage cases. For this calculation, it was conservatively assumed that all activity in the flow paths into the control room (filtered and unfiltered) was captured by the filters. (The activity collected on the filters from the containment pressure relief line is negligible and was not considered). The whole body dose from the activity on the control room filters was calculated using two-dimensional discrete ordinates techniques with the reference dose point location at an axial elevation approximately seven feet below the control room ceiling, approximating the location of personnel. The geometric model used in the transport calculations included the filter structure, the filter support pad, and the control room roof. The iodine activity was assumed to be uniformly distributed on the filter material.

Activity Remaining in Containment

The direct dose to the control room operators from activity dispersed in the containment building was calculated based on the geometric configuration of the Comanche Peak Units using point kernel techniques. The point kernel geometric model included only the cylindrical walls of the building and the hemispherical dome topping the structure. The presence of internal walls was conservatively ignored in the model. The source released to the containment atmosphere was uniformly distributed and was assumed to be reduced only by radioactive decay. Source reduction by other mechanisms such as the action of containment sprays was conservatively ignored in the analysis.

Results

Table 4-1 contains the calculated control room whole body doses due to activity that has entered the control room, the external cloud of activity, and streaming from the external cloud of activity. Table 4-1 shows the breakdown of the doses from the three release paths.

Table 4-1: Control Room Whole Body Doses (rem)

Control Room Model	Containment Leakage	ESF Leakage	Containment Pressure Relief Line
Activity in Control Room	7.4102E-01	9.0470E-04	6.9348E-05
External Cloud	5.2202E-02	5.9889E-03	1.2437E-06
Streaming of External Cloud	1.8218E-01	2.0901E-02	4.3405E-06
Total	**9.7540E-01**	**2.7795E-02**	**7.4932E-05**

The dose due to activity on the control room filters was not calculated for each release path, but rather the activity from all paths was totaled and then the dose was calculated. The total whole body dose due to activity on the control room filters was calculated to be 1.3E-1 rem.

The whole body control room dose due to activity remaining in the containment building was calculated to be 4.0E-2 rem.

For reporting, the doses that were calculated above were increased by 5% for additional conservatism and rounded. Table 4-2 contains the calculated and reported whole body control room doses.

Table 4-2: Reported Control Room Whole Body Dose (rem)

	Calculated	Reported
Containment Leakage	9.7540E-01	1.02E+00
ESF Leakage	2.7795E-02	2.92E-02
Containment Pressure Relief Line	7.4932E-05	7.87E-05
Direct Dose from Containment	4.0E-2	4.20E-02
Dose from Activity on Filters	1.3E-1	1.37E-01
Total		**1.2E+00**

NRC Question 5.

In SPULR Section 2.9.8, "Radiological Consequences of Fuel Handling Accident," it is stated that a fuel decay time of 50 hours was used for the analysis. By what means do you control movement of the fuel prior to 50 hours?

CPNPP Response:

The required fuel decay time prior to fuel movement is governed by Technical Requirements Manual TR 13.9.31, *Decay Time*. This requirement is implemented using station refueling procedure RFO-102.

Currently, the required fuel decay time prior to initiating fuel movement within the reactor vessel is 100 hours, per TR 13.9.31. This requirement is consistent with assumptions of the Fuel Handling Accident (FHA) analysis of record.

For SPU, Comanche Peak has prepared a revision to the FHA analysis assuming shutdown occurs from steady-state operation at SPU conditions. In anticipation of future outage schedule improvements that may reduce the time necessary to prepare for fuel movement, the FHA analysis for SPU was performed assuming fuel decay times of 50 and 75 hours. Revision of Technical Requirements Manual TR 13.9.31 and RFO-102 to allow initiating fuel movement less than 100 hours after shutdown is scheduled to occur prior to SPU implementation under 10CFR50.59 through plant procedures and programs. The actual change is bounded by the assumption of 50 hours reported within the SPU Licensing Report.

CONTAINMENT VENTILITATION BRANCH

NRC Question 1. SPULR Section 2.3.5 "Station Blackout"

Subsections 2.3.5.2.1 and 2.3.5.2.3 provide the conservative maximum temperatures for a four-hour coping period and concludes that there is no impact due to the stretch power uprate (SPU) on equipment required to cope with Station Blackout (SBO). Provide details of how the calculated temperatures compare with pre-SPU conditions and the operability evaluations performed.

CPNPP Response:

Licensing Report Subsection 2.3.5.1 identifies these areas as containing SBO equipment: Uninterruptible power supply (UPS) and distribution rooms, the battery rooms, the switchgear rooms, and the turbine-driven auxiliary feedwater pump area. Heat loads in the UPS and distribution rooms, the switchgear rooms, and the battery rooms do not vary as a function of plant power generation level. The heat sources are from electrical/control equipment and ambient outside air temperatures. None of these sources will measurably increase at SPU. The possible changes in heat loads from power cables are insignificant due to small changes in the current draws. Therefore, the maximum room temperatures would not change as a result of SPU.

In the turbine driven auxiliary feedwater pump area, the heat source is from the steam supply lines to the turbine driven auxiliary feedwater (TDAFW) pump and the steam exhaust lines from the turbine. At SPU conditions, main steam temperature increases by 1.2 °F. However, the heat gain rate (due to higher steam temperature) increases by a ratio of the temperature difference between the steam temperature in the pipe and the ambient room temperature. This ratio is very small and the heat gain rate only increases by a factor of approximately 0.003. The net result is that the room temperature increase would be less than 0.1 °F.

Since the room temperatures are not increasing or are increasing by a negligible amount, operability evaluations are not affected by SPU.

NRC Question 2. SPULR Section 2.6 "Containment Review Considerations"

[A] Verify that all input parameters to the containment peak pressure and temperature, environmental qualification and subcompartment analyses remain the same as those in the updated final safety analysis report except for those affected by the SPU. For example, containment volume, heat sink descriptions, heat exchanger performance, equipment flow rates and flow temperatures, initial relative humidity, ultimate heat sink temperature, etc. Justify any changes made for the SPU analyses.

[B] Subsection 2.6.1.2.3 assumes the same spray efficiency for both injection and recirculation phases for the containment spray system. Explain why the same spray efficiency is used for both injection and recirculation phases of the LOCA when the spray water temperature is much higher during the recirculation phase. Why is spray efficiency used rather than allowing GOTHIC to calculate the heat and mass transfer to the spray drops?

CPNPP Response:

[A] Revisions are planned for the updated final safety analysis report to capture the methodology and the codes that were used for the long term LOCA and steamline break mass and energy releases for the SPU containment integrity analyses with the GOTHIC code. The same standard Westinghouse LOCA mass and energy release methodology from WCAP-10325-P-A [Reference 1] was used to calculate the long-term double-ended pump suction and the double-ended hot leg break releases for the SPU. The Westinghouse methods for calculating the main steamline break mass and energy releases were changed from the LOFTRAN code to the RETRAN code (WCAP-14882-P-A) [Reference 2] for the SPU.

The GOTHIC containment model for Comanche Peak Units 1 and 2 was built in accordance with Westinghouse analysis methods and NRC guidelines and limitations for the use of the DLM model which is a change from the usage of Uchida/Tagami heat transfer correlations in CONTEMPT. The heat sink surface areas, thickness, paint coatings, and material properties were same as the current design analyses of record with the CONTEMPT code. The spray flow rates and actuation setpoints were the same as the current design analyses of record with the CONTEMPT code as was the residual heat removal (RHR) heat exchanger performance. All of the SPU cases with GOTHIC were performed with the highest allowable Technical Specification containment operating pressure of 16.2 psia as the initial containment pressure along with a low initial relative humidity of 15%. This differs from the sensitivity cases with CONTEMPT that used a containment pressure of 14.2, 14.7, and 16.2 psia with an initial relative humidity of 100% because the sensitivities to flashing in GOTHIC differ from CONTEMPT when addressing maximum peak calculated containment pressure. The spray delay times used in the GOTHIC SPU cases correspond to the longest delay time for a loss of offsite power in order to bound cases where offsite power could be available. This reduced the spectrum of cases that needed to be considered for steamline break transients.

All input parameters to the subcompartment analyses remain the same as those in the updated final safety analysis report except for mass and energy rates which are affected by the SPU.

References

1. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version," May 1983.
2. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors for Non-LOCA Safety Analyses," April 1999.

[B] System "efficiency" of 84.2% was developed to account for blocked spray nozzles and pump degradation. The same "efficiency" exists during the injection phase and the recirculation phase but the flow rates that were input to GOTHIC during those phases are different. The containment spray efficiency that is described in Section 2.6.1.2.3 and Table 2.6.1-1 is combined with the design flow rate to arrive at the equivalent spray system flow rate during the injection phase and the recirculation that is input to GOTHIC. In addition, the "efficiency" is essentially a degraded analysis value for the assumed spray flow rate. GOTHIC calculated the heat and mass transfer to the spray droplets.

NRC Question 3. <u>SPULR Section 2.6.2 "Subcompartment Analyses"</u>

Provide the value of pressure differential calculated for the Residual Heat Removal line break within the steam generator (SG) compartment for CPSES, Units 1 and 2, and for the spray line break within the pressurizer cubicle for CPSES, Unit 1 with the Δ76 model SG. Also provide the acceptance criterion for each.

CPNPP Response:

Maximum values of local wall pressure differential calculated for the Residual Heat Removal (RHR) line break within the steam generator compartment for CPNPP, Units 1 and 2 are 14.15 and 13.76 psid, respectively.

Maximum value of local wall pressure differential calculated for the spray line break within the pressurizer cubicle for CPNPP, Unit 1 with the Δ76 model steam generator is 15.64 psid.

Acceptability of the increased calculated pressure differentials is determined by the design margin available in the structural analysis and design of the steam generator compartment and pressurizer cubicle. Review of the design analysis has determined that the affect of increased differential pressures to the total load demand on the affected structures is insignificant with negligible effect on existing design margins.

NRC Question 4. **SPULR Section 2.6.4 "Combustible Gas Control System"**

Is metal-water reaction increased by the SPU? What is its effect on containment response? What, if any, changes are necessary to the hydrogen purge system operation due to the SPU?

CPNPP Response:

In License Amendment 117 to the CPNPP Technical Specifications, the NRC approved CPNPP's application to remove the hydrogen recombiners from the plant design basis. The power uprate activities do not affect the bases for CPNPP's application nor the NRC's Safety Evaluation approving that proposed change. As such, it is not necessary to calculate the hydrogen generation following design basis accidents. However, the heat generated by such metal-water reactions is considered in the containment response to the design basis accidents.

NRC Question 5. **SPULR Section 2.7.4 "Spent Fuel Pool Area Ventilation System"**

Subsections 2.7.4.2.2 and 2.7.4.2.3 indicate that the decay heat in the spent fuel will increase due to the SPU conditions, but the pool water temperatures will remain below pre-SPU design limits. Based on this, it was concluded that the spent fuel pool area ventilation system will maintain the required temperature conditions for personnel and equipment during SPU operation. Are the area coolers designed for the pool temperature at the design limits? Clearly define the areas that will see higher heat loads due to the SPU, the magnitude of the increase, and the basis for determining that the existing system(s) are adequate under post SPU conditions. Also, address whether there are any effects on the ventilation system due to the SPU that could result from loss of SFP cooling.

CPNPP Response:

The Fuel Building ventilation system and it coolers are designed to maintain suitable ambient conditions during normal operations and scheduled shutdowns and are designed for those conditions as stated in the FSAR. During a LOCA the Fuel Building Ventilation system is shut down and cooling to the Pump Area is provided by emergency fan coil units.

The post SPU conditions of operation of SFP cooling system and its impact on SFP areas ventilation has been analyzed by a calculation which concluded that a small increase in the fuel decay heat at SPU is bounded by pre-SPU ventilation system heat loads based on the pool water temperatures remaining below pre-SPU design limits.

A loss of SFP cooling will have no effect on the Fuel Building area ventilation system at SPU conditions. There are no changes required to the Spent Fuel Pool area ventilation system, the Primary ventilation system or the Emergency cooling fan units a result of the SPU or a loss of SFP cooling.

NRC Question 6. <u>SPULR Section 2.7.5 "Auxiliary and Radwaste Area and Turbine Area Ventilation Systems"</u>

[A] Subsections 2.7.5.2.2 and 2.7.5.2.3 discuss the changes in heat loads for the ventilation subsystems in areas served by the auxiliary (Auxiliary Building ventilation) and radwaste area ventilation systems and the turbine area (Turbine Building area ventilation) ventilation systems. The evaluation concludes that the increase in the heat loads caused by the SPU will have no effect on the temperature conditions inside these areas. Clearly define the areas that will see higher heat loads due to the SPU, magnitude of the increase, and the basis for determining that the existing systems are adequate under post SPU conditions.

[B] Subsection 2.7.5.2.3 also includes a statement that "for plant areas that use outside air exchange to provide cooling, outside air temperature changes dominate any potential temperature changes caused by the SPU." Staff requests clarification of this statement. Are the systems designed for specific outdoor and indoor design temperatures? Assuming the calculations were done with steady state conditions, what benefit will outside air temperature changes have on days when the outside air temperature exceeds the selected value?

CPNPP Response:

[A] The Fuel Building ventilation system and it coolers are designed to maintain suitable ambient conditions during normal operations and scheduled shutdowns and are designed for those conditions as stated in the FSAR. During a LOCA the Fuel Building Ventilation system is shut down and cooling to the Pump Area is provided by emergency fan coil units.

The post SPU conditions of operation of SFP cooling system and its impact on SFP areas ventilation has been analyzed by a calculation which concluded that a small increase in the fuel decay heat at SPU is bounded by pre-SPU ventilation system heat loads based on the pool water temperatures remaining below pre-SPU design limits.

A loss of SFP cooling will have no effect on the Fuel Building area ventilation system at SPU conditions. There are no changes required to the Spent Fuel Pool area ventilation system, the Primary ventilation system or the Emergency cooling fan units a result of the SPU or a loss of SFP cooling.

[B] The heat load changes due to SPU in the auxiliary (Auxiliary Building ventilation) and radwaste area ventilation systems and the turbine area (Turbine Building area ventilation) ventilation systems are insignificant. The turbine building areas (Turbine Building area ventilation) that are served by direct ventilation with the outdoor air normally experience room air temperature daily variation corresponding to the changes within the plant outdoor air temperature design limits.

NRC Question 7. <u>SPULR Section 2.7.7 "Other Ventilation Systems (Containment)"</u>

Subsection 2.7.7.2.3 provides the results of the evaluation in terms of an increase in the containment bulk air temperature of less than 0.15°F (Degrees Fahrenheit) from current observed level. Provide details of the analysis or calculations performed to predict the temperature increase based on current observed level. Is it based on calculations different from design basis calculations?

CPNPP Response:

The SPU impact on Containment Structure environment has been evaluated by a calculation for heat release from Feedwater System and Main Steam piping, and from NSSS equipment. The heat gain rate (due to higher steam and feedwater temperatures) increases by a ratio of the temperature difference between the steam/feedwater temperature in the pipes and the ambient room temperature. This ratio is very small and the heat gain rate only increases by a factor of approximately 0.005. The net result is that the Containment Structure temperature increase would be less than 0.15 °F. As such, the SPU calculations are different from the design basis calculations, only in that they scale the pre-SPU temperatures by a factor representing the small effect of uprate.

STEAM GENERATOR INTEGRITY AND CHEMICAL ENGINEERING BRANCH

NRC Question 1.

In SPULR, Section 2.2.2.5.2.1, page 2.2.2-51, it is stated that the stress and fatigue results were calculated by applying scale factors to the previously calculated baseline stress and fatigue values. Describe the scale factor(s) used.

CPNPP Response:

The scale factors for the primary-side components are based on the change in pressure differential resulting from the SPU. Factors are calculated for each transient based on the ratio of pre-uprate to post-uprate pressure differentials and are applied to both the pressure and thermal stress calculated in the baseline analysis. No credit is taken for ratios less than 1.0. Note that the temperature change for the primary-side transients change on the order of one to two degrees Fahrenheit and that this effect is not significant. Therefore, applying a factor based on the change in the pressure differential to the thermal stress is conservative.

For secondary-side components, the change in pressure stress as a result of the uprate is calculated. This change in pressure stress is based on 100% power conditions and is applied to increase the stress range used to calculate the fatigue usage regardless of whether the stress could actually decrease the stress range. For those components where temperature changes could result in a change in the thermal stress, a factor is applied based on the ratio of the temperature change from ambient, pre- and post-uprate. The components affected from temperature changes are those associated with the feedwater system since a change in the feedwater temperature will directly affect the thermal transient across the component and so contributes to the stress range calculated. Other components not directly in contact with the feedwater will not be impacted since the significant thermal stresses are produced during the initiation of the transient and are not affected by the SPU. Again, the stresses are applied in the conservative direction and no credit is taken for any reduction in stress that may result from the SPU.

NRC Question 2.

Confirm that the SPULR SG tube integrity analyses addresses the current condition of your SGs (e.g., plugs, tube repairs, loose parts, etc.). In addition, provide confirmation that your SG tube plugging criteria for CPSES, Units 1 ands 2 is still appropriate for SPU conditions, given the guidance in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR (Pressurized Water Reactor) Steam Generator Tubes."

CPNPP Response:

Tube integrity analyses follow the guidance of the EPRI Integrity Assessment Guidelines. All reported degradation is evaluated in accordance with these guidelines. Units 1 and 2 at CPNPP contain only Alloy 690 shop or field installed welded plugs (prior to commercial operation), or field installed Westinghouse Alloy 690 mechanical (ribbed) plugs (post commercial operation). Inspection of plugs is performed visually in accordance with the latest revision of the EPRI PWR SG Examination Guidelines. As there are no industry reports of Alloy 690 plug degradation as well as no reports at CPNPP, the SG integrity assessment does not address Alloy 690 plugs. CPNPP does not contain tube repairs (i.e., sleeving). Potential tube damage from foreign objects/loose part interaction is evaluated in the condition monitoring stage. All reported tube degradation due to foreign object/loose part interaction is plugged, and/or plugged and stabilized. Foreign objects/loose parts which cannot be retrieved are evaluated for their impact upon SG tube integrity until the next scheduled inspection.

A Regulatory Guide 1.121 analysis has not been performed specifically for CPNPP. The 40% TW repair criterion can be validated using the equations of the EPRI Flaw Handbook. For infinite length, 360 degree uniform thinning Section III of the ASME Code (Equation 5-30 of Flaw Handbook) has been used to define a minimum acceptable wall thickness. This equation is controlled by the ASME Code minimum ultimate strength value and thus applies to both CPNPP units. For an assumed bounding normal operating condition pressure differential of 1333 psi for the 589.2 °F reactor vessel average temperature condition the required minimum wall thickness which satisfies the ASME Code stress limit is 39.5% (60.5% degraded depth). However, uniform thinning of infinite length is not a realistic degradation mode for CPNPP. The only degradation mechanisms applicable to CPNPP which can be sized by eddy current analysis and justified for continued operation are tube wear at support structures. Postulated wear scar lengths would be limited to the maximum axial length of any support structure, or 1.125 inch. For condition monitoring purposes Section 5.3.2 of the EPRI Flaw Handbook would be used to estimate the burst capability of observed degradation against the NEI 97-06 performance criterion.

NRC Question 3.

In SPULR Section 2.1.8, it is indicated that the Flow-Accelerated Corrosion (FAC) program implements the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program." However, the revision of the report is not specified. It is noted by the NRC staff that the latest revision, Revision 3, was issued by EPRI in May 2006 and includes feedwater heaters and other vessels. Describe which heaters and/or vessels are included in the CPSES, Units 1 and 2 FAC program and the results of the last inspection. Also, discuss whether additional inspections will be performed to assess wear prior to entering SPU conditions.

CPNPP Response:

The Flow-Accelerated Corrosion (FAC) Program at CPNPP is in compliance with the EPRI Report NSAC-202L "Recommendations for an Effective Flow-Accelerated Corrosion Program" Revision 2. Beginning in 2001, CPNPP began to implement inspections to feedwater heaters. Since that time all feedwater heaters have been inspected a minimum of one time. All results exceeded minimal required thicknesses. Inspection results are available at CPNPP if required. There are no scheduled FAC inspections on the feedwater heaters between now and the implementation of the SPU conditions.

NRC Question 4.

Section 3.0 of the Spent Fuel Pool Criticality Analysis, WCAP-16827-P (SFPCSA) discusses the proposed RackSaver inserts for the spent fuel pool (SFP). These inserts, which are to be utilized in the CPSES, Units 1 and 2 SFPs, are chevron-shaped and fabricated from two sheets of aluminum-boron carbide metal matrix composite material.

Provide the following additional information for the RackSaver inserts:

- **Qualification test reports.**
- **Details of the fabrication process (e.g. anodizing, welding verses mechanical fasteners, etc.) and the surveillance program for degradation monitoring. Also, provide the acceptance criteria for the surveillance program for the RackSaver inserts.**

CPNPP Response:

Although the Spent Fuel Pool Criticality Analysis, WCAP-16827-P (SFPCSA) discusses the proposed RackSaver inserts for use in the spent fuel pool (SFP), the proposed CPNPP License Amendment Request (LAR) is not requesting use of the RackSaver inserts at this time. CPNPP is only requesting use of WCAP-16827-P for revised enrichment verses burnup curves resulting from the changes to spent fuel reactivity due to operation at SPU conditions (See SPULR Section 2.8.6.2, "Spent Fuel Storage"). Use of the RackSavers inserts described in WCAP-16827-P would require additional changes to the CPNPP Technical Specifications and a subsequent LAR for Units 1 and 2. Based on the above discussion, should the staff still desire to review the RackSaver insert information, it can be made available for review.

BALANCE OF PLANT BRANCH

NRC Question 1:

Section 2.5.1.3.2.2.3 of the SPULR, "Evaluation of Piping Failures," describes that, although there are no changes to the design conditions of the safety injection system due to the SPU, there are insignificant changes in the volumes of fluid or the mass and energy release for the safety injection system break scenarios. Explain the cause of the changes in fluid release and identify any changes in analysis assumptions, methodology, and design inputs between the existing analyses and the analysis performed for SPU conditions.

CPNPP Response:

The safety injection system pressures, temperatures, and flows are not impacted by the uprate. The statement in LR Section 2.5.1.3 that "there are insignificant changes in the volumes of fluid or the M&E release for the above system break scenarios" was meant to refer the group of systems discussed above this paragraph, and in particular the Main Steam and Feedwater Systems which had small changes. As stated, there are no changes to the design conditions of the Safety Injection System due to SPU.

NRC Question 2:

Section 2.5.1.3.2.2.4 of the SPULR, "Summary of Impact on Building Environments," describes that the existing flooding volume for feedwater pipe breaks is still bounding for SPU. Identify any changes in analysis assumptions, methodology, and design inputs between the existing flooding volume analysis and the analysis performed for SPU conditions.

CPNPP Response:

The flooding analyses for Unit 1 and Unit 2 receive an input from the high energy line break (HELB) analyses for break flow rate corresponding to a one square foot non-mechanistic break in the feedwater line.

The existing HELB analyses for the non-mechanistic break of the feedwater line in the penetration exclusion area apply a very conservative approach in determining break flow rate. This approach considers choke flow conditions and maximum feedwater pump flow conditions.

Values for feedwater system pressure and temperature have increased slightly in Unit 1 and Unit 2 at SPU. This would lead to a slightly higher contribution to the break flow from the feedwater pumps. Steam generator pressures at no load conditions are essentially unchanged for Unit 1 and show a slight decrease for Unit 2 at SPU. With a slight decrease in steam generator pressure, the break flow contribution from the steam generator would decrease slightly for Unit 2.

No changes have been made to the analysis assumptions and methodology as a result of SPU.

With small changes in feedwater system pressure and temperature and small changes in steam generator pressure, the changes in break flow rate are expected to be small.

The break flows determined by the existing methodology with conservative assumptions provide a conservative value that would bound small changes to break flow rate at SPU conditions. Since the existing break flow rates would still be bounding, the flooding volumes in the existing analyses would still be bounding and the flooding calculations remain unchanged.

NRC Question 3:

Section 2.5.2.2.2 of the SPULR, "Description of Analyses and Evaluations," describes that the pressurizer relief tank (PRT) design is conservatively sized to condense and cool a steam discharge equal to 105 percent of the full-power pressurizer steam volume, and that the loss of external electrical load transient analysis determined that the pressurizer steam mass and energy discharged into the PRT is less than the design basis discharge. However, the CPSES, Units 1 and 2 Final Safety Analysis Report (FSAR), through Amendment 101, states that the PRT is sized to receive and condense a discharge of 110 percent of the full-power pressurizer steam volume and that this steam volume requirement is approximately that which would be experienced if the plant were to suffer a complete loss of load accompanied by a turbine trip, but without the resulting reactor trip. Clarify the correct portion of the full-power pressurizer steam volume the PRT is sized to receive and how the loss of external electrical load transient analysis described in the SPULR compares with the event described in the FSAR.

CPNPP Response:

The original PRT design was based on the requirement to receive and condense a discharge of steam equivalent to 110% of the full power pressurizer steam volume. This was based on a conservative approximation of the non-design basis event described in FSAR section 5.4.11.1. The current design basis for the PRT evaluated for the SPU is to receive the equivalent of 105% of the full power pressurizer steam volume. This 105% volume is conservative with respect to the steam discharge for the current SPU Chapter 15 loss of load analysis.

FSAR section 5.4.11.1 will be updated per plant procedures and programs to reflect this evaluation.

NRC Question 4:

Section 2.5.4.1 of the SPULR, "Spent Fuel Pool Cooling and Cleanup System," describes that, although there are no modifications to the design of the spent fuel pool cooling and cleanup system related to the SPU, the system is capable of maintaining the pool temperature limit at a total heat load of 57.9 million British thermal units (BTU) per hour at 150 hours after shutdown, which is more challenging to the cooling system than the total heat load of 56.3 million BTU/hr at 168 hours after shutdown listed in Section 9.1.3 of the CPSES, Units 1 and 2 FSAR. Explain how the temperature limit is maintained at the higher heat load and identify any changes in analysis assumptions, methodology, and design inputs between the existing analyses described in the FSAR and the analysis performed for SPU conditions.

CPNPP Response:

The calculation that determined the pool temperature was revised for SPU heat load values. The methodology and analysis assumptions remain the same. It has been calculated that the Spent Fuel Cooling System (SF) has the capability to continue to cool the pools and keep them within the temperature limit. The SF cooling pump alignment to the fuel pools is unchanged for SPU. The cooling system is aligned so that one pump is connected to the pool with the maximum heat load and the other pump is connected to both pools, maintaining both spent fuel pools at or below 150 °F.

NRC Question 5:

Section 9.2.2.3 of the CPSES, Units 1 and 2 FSAR describes that valves CC-0107, CC-0109, CC-0157, and CC-0158 have modified discs which have been drilled to serve as flow restriction orifices. These valves are closed to provide acceptable component cooling water (CCW) flow balancing for design basis accidents to limit heat addition to CCW. Describe how the heat addition to the CCW system at

SPU design basis accident conditions has been evaluated to demonstrate that the CCW temperature is maintained at 135°F or below during containment spray recirculation with a service water system temperature of 102°F, consistent with the discussion in Section 2.5.4.3.2.3 of the SPULR. Also, clarify the basis for concluding the heat addition from normal shutdown cooling at a CCW temperature of 122°F is more limiting than the post-accident condition.

CPNPP Response:

Valves CC-0107 and CC-0158 are in the CCW supply line to the '01' and '02' containment spray heat exchangers, respectively, while valves CC-0109 and CC-0157 are in the CCW supply lines to the '01' and '02' residual heat removal heat exchangers (Ref. FSAR Figure 9.2-3). Each of the four valves is maintained in the closed position during Modes 1, 2, and 3 and none of the valves are required to be repositioned in order to perform their respective safety functions to mitigate a design basis accident. The disc of each of these valves has been drilled so that, when in the normal, closed position, they act effectively as orifice plates and balance CCW flow to various system loads during a postulated accident.

During a postulated LOCA, the non-safeguards loads normally cooled by CCW are automatically isolated. While realignment of the CCW system in response to a safety injection signal does result in a change in system flow balance, the resultant flow balance ensures fulfillment of safety functions. Procedural guidance ensures the maximum system temperature limit of 135 °F is not violated throughout the event sequence. The safeguards heat loads placed on the system can be considered in two groups (1) time-dependent heat loads, such the RHR and CSS heat exchangers, and (2) time-independent or constant loads such as control room air conditioners, etc. The time-independent loads are unaffected by a change in power level. Thus, the total effect of SPU on post-LOCA heat removal via the CCW heat exchangers can be characterized as a change in heat removal via the RHR and CSS heat exchangers during the recirculation phase of the event, only.

The maximum allowable bulk temperature at the outlet of each CCW heat exchanger following a LOCA is 135 °F and has remained unchanged following SPU. For purposes of verification the post-LOCA CCW heat exchanger outlet temperature did not exceed the system limit. The analysis assumed maximum RHR and CSS flow rates, minimum CCW system flow, including minimum CCW flow to the RHR and CSS heat exchangers, zero fouling of the RHR and CSS heat exchangers, as well as a conservatively high sump temperature. The results of the analysis verify the CCW temperature limit (135 °F) would not be exceeded.

As post-LOCA CCW system operation would be based on the maximum 135 °F heat exchanger exit temperature limit, variations in other parameters, including SSI temperature, RHR flow rates, etc. would only result in a change in the rate of heat transfer to the SSI from the accident unit. In the analysis of postulated LOCA events, variations in the rate of heat removal from the sump fluid, through the CCW system (via RHR and CSS heat exchangers), into the Service Water System and ultimately to the SSI are manifest by variations in the containment sump time-temperature transients. That is, parameter variations which decrease heat removal capability, i.e., postulation of a single failure of one train of cooling, result in an increase in the time necessary to reduce containment temperature.

The peak sump temperature occurs prior to the initiation of the recirculation phase and hence is unaffected by any of the considerations discussed above.

The following discussion clarifies the basis for concluding that the heat addition from normal shutdown cooling at a CCW temperature of 122 °F is more limiting than the post-accident condition (consistent with FSAR Section 4.3).

The design and licensing cooldown rate requirements (i.e., the time during which the Reactor Coolant System [RCS] must be cooled from one temperature to another) are more severe for a cooldown than for the LOCA.

During normal shutdown cooldown of the RCS, the CCW system must meet requirements for cooldown heat load removal as well as requirements for heat load removal from other plant equipment (i.e., both safeguards and non-safeguard heat loads). During post accident operation, non-safeguards heat loads are isolated. As a result, the heat load removal rate during cooldown operation is substantially greater than during post accident operation.

The maximum allowable CCW temperature during cooldown is 122 °F as compared with 135 °F for LOCA. Assuming maximum SW/SSI system temperature of 102 °F, this results in the significantly smaller ΔT of 20 °F (122 °F - 102 °F) between the CCW system and the SW/SSI system for the cooldown case as compared with the ΔT of 33 °F (135 °F - 102 °F) for the LOCA case. However, the total CCW heat transfer rate is greater for the cooldown case (i.e., considering both the safeguard heat loads and the non-safeguards heat loads) than for the LOCA case since there is more CCW flow required during cooldown than during LOCA. By analysis, the "Q" (in Btu/hr) for the cooldown case is substantially greater than the "Q" for the LOCA case.

NRC Question 6:

Section 10.4.9 of the CPSES, Units 1 and 2 FSAR states that the auxiliary feedwater (AFW) system is capable of supplying the minimum required flow to at least two of the effective SGs against a back pressure equivalent to the accumulation pressure of the lowest set safety valve (1236 pounds per square inch absolute) plus the system frictional and static losses, and Section 2.5.5.1 of the SPULR states that the setpoints for the main steam safety valves would be unchanged for SPU operation. In addition, the Bases for Technical Specification (TS) 3.7.5 state that a single motor driven AFW pump provides 100 percent of the required system capacity. Identify the most limiting event with respect to AFW flow capacity and address whether a single motor driven AFW pump would continue to provide 100 percent of the required capacity at the increased decay heat rate associated with the SPU.

CPNPP Response:

(1) Feedline Break

- The single failure assumption was conservatively set as the loss of a motor-driven AFW pump (MDAFW) feeding two intact steam generators; all the flow from the second motor-driven AFW pump is assumed to flow out of the break along with a portion of the turbine-driven AFW (TDAFW) flow until the faulted steam generator is isolated.
- For the first 30 minutes following reactor trip, a total of 430 gpm of AFW flow from the turbine-driven AFW pump was split equally among the three intact steam generators. Following isolation of the faulted steam generator, an additional 370 gpm was made available to split among the 3 intact steam generators. No AFW flow is assumed to reach the faulted steam generator following isolation. AFW flow from the second motor-driven pump was conservatively not modeled.

(2) Loss of Normal Feedwater

- It was assumed that two MDAFW pumps are available to supply flow to all four steam generators, 60 seconds following a low-low steam generator water level signal. The worst single failure for this analysis is the loss of the TDAFW pump.

These assumptions are consistent with the current analysis supporting the CPNPP FSAR.

The bases section of TS 3.7.5 will be updated per plant procedures and programs to reflect the above assumptions. The statement that a single-motor driven AFW pump will provide 100 percent of the required capacity will be changed to reflect the current accident analyses.

NRC Question 7:

Section 10.3.2.2 of the CPSES, Units 1 and 2 FSAR states that the capacity of each Atmospheric Relief Valve is sufficiently large to allow the plant to be cooled from no-load temperature to the residual heat removal system cut-in temperature of 350_°F prior to the time that the condensate storage tank is exhausted. Describe how this design capability was confirmed for the increased decay heat rate associated with the SPU.

CPNPP Response:

The Atmospheric Relief Valves (ARVs) at CPNPP were originally sized to accommodate the Engineered Safeguards Features (ESF) rating of the plant, to permit the cooldown to be completed with the capacity of the normally aligned water source (Condensate Storage Tank). The ESF rating at Comanche Peak is an NSSS power level of 3635.8 MWt (104.5% of the original core power rating of 3411 MWt, plus 2% for measurement uncertainty). Net RCP heat is not included in this total since the cooldown scenario has no RCPs running. The analogous NSSS power level at the proposed power uprate level is approximately 3633.7 MWt (3612 MWt plus 0.6% uncertainty). The original sizing basis of the ARVs was confirmed to be greater than the uprate sizing basis of the ARVs: therefore, a detailed calculation was not performed for the SPU.

Note that the FSAR statement (Section 10.3.2.2) mentioned in the above question is based on 2 ARVs being available. This clarifying phrase will be added to FSAR Section 10.3.2.2 per plant procedures and programs.

NRC Question 8:

Section 11.3.2.1.1 of the CPSES, Units 1 and 2 FSAR States that a catalytic recombiner recombines hydrogen brought into the Gaseous Waste Management System (GWMS) with a controlled amount of oxygen to form water. The control system for the recombiner maintains an oxygen lean mixture to preclude the possibility of a hydrogen explosion. Describe how the SPU affects the ability to preclude creation of potentially explosive gas mixtures in the GWMS.

CPNPP Response:

The SPU will not affect the ability of the of the recombiner control system to maintain an oxygen lean mixture. The system processes the hydrogen vented or removed from the reactor coolant measuring both the hydrogen and the oxygen content of the gas entering the recombiner loop and adjusting the oxygen addition to maintain a lean mixture in the combiners.