January 14, 2010

Mr. Randall K. Edington
Executive Vice President, Nuclear
Mail Station 7602
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3, LICENSE RENEWAL APPLICATION (TAC NOS. ME0254, ME0255, ME0256)

Dear Mr. Edington:

By letter dated December 11, 2008, as supplemented by letter dated April 14, 2009, Arizona Public Service Company (APS) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew Operating License Nos. NPF-41, NPF-51, and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The staff of the U.S. Nuclear Regulatory Commission is reviewing this application and has identified, in the enclosure, areas where additional information is needed to complete the review. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with APS staff on December 23, 2009 and January 12, 2010, and a mutually agreeable date for your response was determined to be 30 calendar days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-1906 or by e-mail at Lisa.Regner@nrc.gov.

Sincerely,

/RA/

Lisa M. Regner, Senior Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-528, 50-529, and 50-530

Enclosure:
As stated

cc w/encl: See next page

January 14, 2010

Mr. Randall K. Edington
Executive Vice President, Nuclear
Mail Station 7602
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3, LICENSE RENEWAL APPLICATION (TAC NOS. ME0254, ME0255, ME0256)

Dear Mr. Edington:

By letter dated December 11, 2008, as supplemented by letter dated April 14, 2009, Arizona Public Service Company (APS) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew Operating License Nos. NPF-41, NPF-51, and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The staff of the U.S. Nuclear Regulatory Commission is reviewing this application and has identified, in the enclosure, areas where additional information is needed to complete the review. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with APS staff on December 23, 2009 and January 12, 2010, and a mutually agreeable date for your response was determined to be 30 calendar days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-1906 or by e-mail at Lisa.Regner@nrc.gov.

Sincerely,

/RA/

Lisa M. Regner, Senior Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-528, 50-529, and 50-530

Enclosure:
As stated

cc w/encl: See next page

DISTRIBUTION:
See next page

ADAMS Accession No.: **ML093290298**

OFFICE	PM:RPB2:DLR	LA:DLR	BC:RPB1:DLR	PM:RPB2:DLR
NAME	LRegner	SFigeuroa	DWrona	LRegner (Signature)
DATE	12/13/09	12/1/09	01/14/10	01/13/10

OFFICIAL RECORD COPY

Letter to Randall K. Edington from Lisa M. Regner dated January 14, 2010

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3, LICENSE RENEWAL APPLICATION (TAC NOS. ME0254, ME0255, ME0256)

DISTRIBUTION:

HARD COPY:
DLR RF

E-MAIL:
PUBLIC
RidsNrrDlr Resource
RidsNrrDlrRpb1 Resource
RidsNrrDlrRpb2 Resource
RidsNrrDlrRarb Resource
RidsNrrDlrRasb Resource
RidsNrrDlrRapb Resource
RidsNrrDlrRpob Resource
RidsNrrDciCvib Resource
RidsNrrDciCpnb Resource
RidsNrrDciCsgb Resource
RidsNrrDraAfpb Resource
RidsNrrDraApla Resource
RidsNrrDeEmcb Resource
RidsNrrDeEeeb Resource
RidsNrrDssSrxb Resource
RidsNrrDssSbpb Resource
RidsNrrDssScvb Resource
RidsOgcMailCenter Resource
RidsOpaMail Resource

L. Regner
D. Drucker
R. Hall
B. Mizuno, OGC
R. Treadway, RIV
G. Pick, RIV

PALO VERDE NUCLEAR GENERATING STATION
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION

Section 4.3.2.4 *Pressurizer and Pressurizer Nozzles*

RAI 4.3.2.4-1. Page 4.3-39 of the license renewal application (LRA) discusses the effect of Combustion Engineering Infobulletin 88-09. The applicant states that the revised 40-year design basis cumulative usage factor at the worst location (pressurizer bottom head support skirt) is 0.7223. However, the applicant did not discuss the cumulative usage factor at the worst location for the 60-year plant life and whether the fatigue analysis of the bottom head support skirt is a time-limited aging analysis (TLAA). (a) Provide the cumulative usage factor for the worst location at the pressurizer for the period of extended operation or justify how the bottom head support skirt satisfies the allowable cumulative usage factor of the American Society of Mechanical Engineers (ASME) Code, Section III, at the end of the 60-year plant life. (b) Discuss whether the fatigue analysis of the pressurizer bottom head support skirt is a TLAA.

RAI 4.3.2.4-2. Page 4.3-39 of the LRA discusses two flaws detected in the pressurizer support skirt forging weld. (a) The applicant stated that the fatigue crack growth analysis predicted growth from the as-found size of 0.59 inch to 0.6921 inch over the design life. Discuss how many years are assumed for the design life and are assumed in the fatigue crack growth calculation. (b) If the 0.6921 inch flaw size is not calculated for the 60-year plant life, calculate the final crack size at the end of 60 years or justify the structural integrity of the support skirt forging weld to the end of the 60 years.

RAI 4.3.2.4-3. Page 4.3-39 of the LRA states that power uprate and steam generator replacement have no effect on the design reports for any of the three units. (a) Reference the design reports associated with pressurizer and pressurizer nozzles that have been reviewed to determine the impact of power uprate and steam generator replacement. Describe them briefly in the context of Section 4.3.2.4. (b) Clarify whether the loadings on the pressurizer and pressurizer nozzles are affected by the power uprate and steam generator replacement.

RAI 4.3.2.4-4. Page 4.3-39 of the LRA, last paragraph, states that the original stress and fatigue analysis of the surge nozzle have been superseded by the reanalysis for a compressive overlay, which included the thermal stratification and insurge-outsurge effects. Submit the reanalysis. Alternatively, reference the reanalysis and describe the analysis input, method, results, and acceptance criteria in detail, demonstrating that the structural integrity of the surge nozzle will be maintained to the end of 60 years.

RAI 4.3.2.4-5. Page 4.3-40 of the LRA discusses the fatigue crack growth in the original heater sleeve attachment welds. (a) Discuss the postulated initial cracks in the original sleeve-to-inner-wall attachment welds. (b) Discuss the projected final flaw size for the postulated cracks at the end of 60 years. (c) Discuss the allowable flaw size. (d) Discuss the results and describe the methodology used in the “subsequent” report and “code design” reports that are mentioned in the second paragraph on page 4.3-40. (e) Provide the references of these reports.

RAI 4.3.2.4-6. Page 4.3-40 of the LRA, first paragraph, states that for the half nozzle repair method, the original sleeve-to-inner wall attachment welds (i.e., J-groove welds) are analyzed for 60 years and, therefore, are not a TLAA. However, other welds and parts were used for the half nozzle repair. For example, welds were used to join the half nozzle to the weld pad. Discuss why the fatigue analysis of the half nozzles and associated new attachment welds, and weld pads are not considered as a TLAA and not discussed in this subsection.

RAI 4.3.2.4-7. In the LRA, page 4.3-40, 5^{th} paragraph states, “The analysis of the weld pads does not explicitly supersede the results of the fatigue analysis with the tapped anchor holes. Therefore, both fatigue analysis results apply.” (a) Discuss whether the anchor holes and weld pads on the pressurizer bottom are a TLAA. (b) Clarify the above two statements in the context of TLAA. (c) Clarify whether the cumulative usage factors for the anchor bolts and weld pads for 60 years are within the allowable cumulative usage factor of 1.0.

RAI 4.3.2.4-8. In the LRA, page 4.3-42, 5^{th} paragraph states that because the pressurizer heaters were subjected to a high temperature of 779 degrees F for 3,700 hours; the evaluation of the creep effects is not a TLAA. Explain why the evaluation of the creep effect is not a TLAA when heaters are subject to a temperature of 779 degrees F for 3,700 hours. Reference technical paper(s) or report(s) to support your technical basis.

RAI 4.3.2.4-9. In the LRA, page 4.3-42, states, “However, overheating did affect the code fatigue analysis.” Clarify the intent of this statement in the context of the TLAA for the pressurizer heaters.

RAI 4.3.2.4-10. In the LRA, page 4.3-42, last paragraph and page 4.3-43, first paragraph, discuss fatigue analysis revisions due to pressurizer nozzle overlays. The applicant states that the fatigue crack growth analyses do not support safety determinations for a defined design lifetime and are therefore not TLAAs. The applicant stated further, “However, the revised fatigue analyses of the adjacent materials affected by the overlays are time-dependent and are TLAAs unless successfully projected to the end of the period of extended operation.” (a) Clarify whether the cumulative fatigue usage factor calculation and fatigue crack growth calculation were calculated for the adjacent materials to the end of 60 years. (b) Submit the revised fatigue analyses for the adjacent materials or describe in detail the analysis input, methodology, acceptance criteria, and result. (c) Identify all the pressurizer nozzles in all three units that have been weld overlaid and identify “the adjacent materials” affected by the weld overlays. (d) The applicant states that the adjacent materials analyses are TLAA. Discuss the actions that will be taken as a result of the TLAA determination.

RAI 4.3.2.4-11. In the LRA, page 4.3-43, the second and third paragraphs discuss the fatigue usage factors for the surge and spray nozzles for a 60-year life as being 1.44 and 1.49, respectively. The applicant states that the surge nozzle is monitored for fatigue usage and the fatigue usage factor will not exceed the code limit of 1.0 as long as the number of applied load cycles does not exceed the number specified by the design specification for this nozzle and used in the analyses. (a) Clarify whether the spray nozzle will be monitored for fatigue usage. (b) Discuss why a plastic analysis was not performed in accordance with NB-3228 of the ASME Code, Section III when the cumulative usage factors for surge and spray piping exceed 1.0 which were calculated by the elastic analysis of NB-3222. Section 4.3.2.9 discussed a plastic analysis performed by Combustion Engineering on the surge line that lowered the cumulative usage factor.

RAI 4.3.2.4-12. Table 4.3-7 of the LRA summarizes the results of fatigue usage factors. Discuss the actions that will be taken if an item requires the TLAA as shown in the “TLAA” column.

RAI 4.3.2.4-13. In the LRA, Table 4.3-7, item 2, provides no cumulative usage factors for the pressurizer support skirt forging weld. It is stated that crack growth of the flaw in the skirt forging weld is less than the critical size for the assumed number of lifetime cycles. Clarify whether the “lifetime” cycles implies that the transient cycles used in the fatigue analysis are for 40 years or 60 years of plant life. If crack growth of the flaw in the skirt forging weld is calculated based on the 40 year cycles, perform a crack growth calculation using the 60 year cycles or justify why the 40 year crack growth results are acceptable for the period of extended operation.

RAI 4.3.2.4-14. On page 4.3-47 of the LRA, under the Disposition section, the applicant identified the Metal Fatigue of Reactor Coolant Pressure Boundary Program as the program to manage the crack in the support skirt forging weld. There are many pressurizer components listed in Table 4.3-7 that have been identified as requiring TLAA and should be required to be monitored by the metal fatigue aging management program. However, these pressurizer components are not identified in the Disposition section on page 4.3-47. Explain why the affected components in Table 4.3-7 are not included in the Disposition Section on page 4.3-47.

RAI 4.3.2.4-15. In the LRA, page 4.3-47, “Fatigue Analysis,” states that the Metal Fatigue of Reactor Coolant Pressure Boundary program will ensure that the fatigue usage factors based on those transient events that will remain within the ASME Code limit of 1.0 for the period of extended operation or will ensure that appropriate reevaluation or other of corrective actions before the design basis number of these events is exceeded. Identify the exact pressurizer components that will be reevaluated in the context of Section 4.3.2.4.

Section 4.3.2.7 *ASME III Class 1 Piping and Piping Nozzles*

RAI 4.3.2.7-1. On page 4.3-56 of the LRA, the applicant found that reactor coolant system (RCS) piping, nozzles, RTD thermowells, and other Class 1 piping satisfy the current licensing basis design number of transients under power uprate and steam generator replacement. However, the applicant did not discuss whether the piping components have been analyzed for 60 years. (a) Clarify whether these piping components satisfy the allowable cumulative usage factor of 1.0 using a projected number of transients at the end of 60 years. (b) If this calculation has not been performed, perform a fatigue usage factor analysis for 60 years, or justify how the cumulative fatigue usage factors for all Class 1 piping and associated components satisfy the allowable of 1.0 per the ASME Code, Section III at the end of 60 years.

RAI 4.3.2.7-2. In the LRA, page 4.3-56, last paragraph, states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will calculate stress-based fatigue in the chemical and volume control system (CVCS) charging nozzle. Describe the analysis in detail, such as the analysis input, analytical procedures and method, acceptance criteria, and results.

RAI 4.3.2.7-3. Page 4.3-57 of the LRA discusses the reduced wall thickness in the RCS hot leg and cold leg piping. (a) Clarify why the fatigue usage factor for the RCS hot and cold leg piping with reduced wall thickness was not calculated for 60 years. (b) List all Alloy 82/182 welds in the RCS hot leg and cold leg piping. (c) Discuss whether there are any indications/flaws detected in the Alloy 82/182 welds that remained in service. (d) If there are flaws in the Alloy 82/182 welds, perform a fatigue crack growth analysis for the 60-year plant life, or justify why flaw evaluations are not need to demonstrate the structural integrity of the affected welds at the end of 60 years.

RAI 4.3.2.7-4. In the LRA, page 4.3-57, last paragraph, states that the original RCS hot legs contained a total of 27 Alloy 600 small-bore nozzles in each unit. The applicant replaced seven pressure differential transmitter (PDT) nozzles and one sample nozzle in Unit 2 with full nozzles during 1992. On page 4.3-58 of the LRA, the applicant states that the remaining hot leg small-bore nozzles were replaced with the Alloy 690 half-nozzle design. However, it is not clear whether these remaining nozzles are located in Unit 1, 2, or 3 and it is not clear why small-bore nozzles in Units 1 and 3 are not discussed in this section. (a) Provide a table, similar to Table 4.3-7, with the following information: list all 27 small-bore nozzles for each unit, identify the type of the nozzle (e.g., Resistance Temperature Detector (RTD), PDT) or systems, identify the repair method for each nozzle, identify whether a fatigue analysis was performed for 60 years for each nozzle, and specify whether a TLAA is needed. (b) If a nozzle is not analyzed for 60 years, perform a fatigue analysis for 60 years, or justify why a fatigue analysis is not needed to demonstrate that that small-bore nozzle satisfy the ASME Code Section III allowable usage factor of 1.0 at the end of 60 years. (c) Discuss whether cold leg piping contains small-bore Alloy 600 nozzles, whether they were replaced with Alloy 690 nozzles, and whether their fatigue usage factors were analyzed for 60 years.

RAI 4.3.2.7-5. In the 4th paragraph on page 4.3-58 of the LRA, the applicant discusses that the PDT and sampling half-nozzle repairs do not need fatigue analysis of NB-3222.4(d) of the ASME Section III. However, in the 6th paragraph, the welded plugs for the RTD nozzles repairs were analyzed for fatigue per NB-3222.4(e). Explain why a fatigue analysis does not need to be performed for the half nozzle repair, but one is required for the weld plug repair.

RAI 4.3.2.7-6. In the LRA, page 4.3-59, “Redesigned Reactor Coolant System Thermowells,” states that the thermowells modification did not affect the previous conclusion concerning fatigue of the thermowells and that there is no safety determination based on the plant life for these high-cycle loads and therefore no TLAA. (a) Explain why this issue is not a TLAA because the thermowells experience high-cycle fatigue which is time-dependent. (b) Perform a fatigue analysis of the thermowells for 60 years, or justify why a fatigue usage factor analysis for 60 years is not needed to demonstrate that the new thermowell design satisfy the allowable cumulative usage factor of 1.0.

RAI 4.3.2.7-7. In the LRA, pages 4.3-59 and 4.3-60, “Removal of Reactor System Safety Injection Nozzle Thermal Sleeves,” conclude that the modification did not affect the previous conclusion concerning fatigue of the safety injection nozzles. However, the applicant did not discuss whether the fatigue analysis of the safety injection nozzles was based on 40-year or 60-year transient cycles. (a) Perform a fatigue analysis for 60 years, or justify why a fatigue analysis of the safety injection nozzles for the end of 60 years is not needed to demonstrate that the cumulative usage factors of the subject nozzles at the end of plant life satisfies the ASME Code Section III allowable of 1.0. (b) Discuss whether this issue is a TLAA.

RAI 4.3.2.7-8. In the LRA, page 4.3-60, “Flow Stratification Thermal Gradient in the Auxiliary Spray Line and Tee,” states that the cumulative fatigue usage factor, including the effects of thermal gradient in the auxiliary spray line meets ASME Section III for a 40-year plant life.

(a) Perform a fatigue analysis for 60 years, or justify why a fatigue analysis is not needed for 60 years to demonstrate that the cumulative usage factor of the auxiliary spray line satisfies a 60-year plant life. (b) Discuss whether this issue is a TLAA.

RAI 4.3.2.7-9. LRA page 4.3-61, first paragraph. Explain why fracture mechanics analyses of the hot leg surge and shutdown cooling nozzles overlaid by the weld repair are not a TLAA.

RAI 4.3.2.7-10. In the LRA, page 4.3-61, Disposition Section, states, “The Metal Fatigue of Reactor Coolant Pressure Boundary program will continue to confirm that this is so, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached.” (a) Specify the exact piping components and systems that will be monitored under the Metal Fatigue program in the context of Section 4.3.2.7. (b) Discuss how often the actions (e.g., monitoring the transient cycles and review the records) under the Metal Fatigue program will be performed. Section 4.3.1 discusses that a FatiguePro computer software program is used at Palo Verde to monitor the transient cycles; however, it is not clear how often the monitoring is performed by the applicant and when corrective actions are taken.

Section 4.3.2.9 *Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification*

RAI 4.3.2.9-1. In the LRA, page 4.3-64, last paragraph, states that the cumulative usage factor for the surge line is 1.65 using the elastic analysis. Combustion Engineering performed a plastic analysis which reduced the limiting cumulative usage factor to 0.937. (a) Submit both the elastic and plastic analyses. Alternatively, describe the analyses in detail, including methodology, input, acceptance criteria, and results. (b) Clarify whether the fatigue usage factor analysis is based on a 40-year period or 60-year period. (c) On Page 4.3-65 of the LRA, the applicant states that the Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program will be used to monitor the surge line. Discuss whether the Metal Fatigue aging management program will initiate actions based on the elastic analysis result (CUF of 1.65) or plastic analysis result (CUF of 0.937).

RAI 4.3.2.9-2. In the LRA, page 4.3-65, first paragraph, discusses the surge line elbow and risk-informed inservice inspection program. (a) Confirm that the surge line elbow is a component that requires a nondestructive examination to be performed under the inservice inspection program. (b) Discuss how often the surge line elbow will be inspected in each of the 10-year inservice inspection intervals through the 6^{th} inservice inspection interval. (c) Discuss the nondestructive examination method that will be used for each inspection.

RAI 4.3.2.9-3. In the LRA, page 4.3-65, Disposition section, states that the surge line is subject to stress-based fatigue monitoring under the Metal Fatigue of Reactor Coolant Pressure Boundary program. Discuss how often the record of the worst case cumulative usage factors will be reviewed and evaluated by the applicant.

Section 4.3.2.15 *Absence of TLAAs in Fatigue Crack Growth Assessments and Fracture Mechanics Stability Analyses for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures*

RAI 4.3.2.15-1. In Section 3.6.3 of the Standard Review Plan (NUREG-0800), paragraph III.10 states, “The reviewer should determine that the candidate piping does not have a history of fatigue cracking or failure. An evaluation to ensure that the potential for pipe rupture due to thermal and mechanical induced fatigue is unlikely should be performed.”

The technical basis for the leak-before-break (LBB) approval for Palo Verde is provided in a Combustion Engineering (CE) topical report entitled “Leak Before Break Evaluation of the Main Coolant Loop Piping of a CE Reactor Coolant System,” provided as an attachment to a letter dated June 14, 1983 (also, CE report “Leak Before Break Evaluation of the Main Coolant Loop Piping of a CE Reactor Coolant System,” Revision 1, provided as an attachment to a letter dated December 23, 1983). Section 3 of this report describes fatigue calculations to demonstrate the acceptability of fatigue crack growth for various postulated flaws, which demonstrates that fatigue is not an active degradation mechanism of concern. One of these calculations, for a relatively small flaw of 1 inch in depth and 8 inches in length, demonstrates that the crack will not penetrate through wall for a very large number of cycles, principally heat-up and cool-down cycles.

Although the technical basis report for approval of LBB for Palo Verde incorporates fatigue crack growth calculations that have a time-basis (40 years) and/or consider numbers of cycles in the calculations, the first paragraph on page 4.3-72 of the LRA states that the LBB fatigue crack analyses are not TLAAs because the postulated fatigue cracks grow slowly and the fatigue evaluation does not depend on the design life.

Provide an assessment of the acceptability of the fatigue-based aspects of the Palo Verde LBB approval, consistent with the requirements of 10 CFR 54.21(c).

RAI 4.3.2.15-2. Page 4.3-71 of the LRA. (a) List the piping systems in each of three units that have been approved for LBB application. (b) List references of the LBB analyses.

RAI 4.3.2.15-3. Nickel-based Alloy 600/82/182 material in the pressurized water reactor environment has been shown to be susceptible to primary water stress corrosion cracking (PWSCC). (a) Identify any Alloy 82/182 weld metal and Alloy 600 components used in the LBB-approved piping for both units. (b) If LBB piping contains Alloy 600/82/182 material, discuss any measures (such as weld overlays or mechanical stress improvement) that have been or will be implemented to reduce the susceptibility of PWSCC in the LBB piping components. (c) Discuss the inspection history and future inspection frequency of the Alloy 81/182 dissimilar metal butt welds.

RAI 4.3.2.15-4. (a) Discuss the inspection history and results of the LBB-approved piping. (b) If indications or flaws remain in inservice LBB piping, discuss how the indications and flaws will be monitored to the end of the period of extended operation. (c) Discuss future inspection schedules for each of the LBB pipes (other than existing indications and flaws).

RAI 4.3.2.15-5. Discuss whether the LBB analyses have been re-evaluated to determine the impact of operating conditions due to system modifications such as power uprates or steam generator modifications on the LBB analyses for the period of extended operation.

Section 4.7.4 *Fatigue Crack Growth and Fracture Mechanics stability Analyses of Half-Nozzle Repairs to Alloy 600 Material in Reactor Coolant Hot Legs: Absence of a TLAA for Supporting Corrosion Analyses*

RAI 4.7.4.-1. In the LRA, page 4.7-8, second paragraph, states that the applicant made a commitment to monitor the cold shutdown conditions against the assumptions made in the corrosion analysis to assure that the allowable bore diameter is not exceeded over the life of the plant for the second, third, and fourth 10-year inspection intervals. Discuss whether the same commitment will be implemented in the fifth and sixth inspection intervals. If not, provide justification.

RAI 4.7.4.-2. In the LRA, page 4.7-8, "Extension to All Hot Leg Small-bore Nozzles," states that 63 previously repaired small-bore hot leg nozzles in all three units were added to Relief Request 31. The applicant states further that there are a total of 27 small diameter hot leg penetrations per unit. If there are 27 small-bore nozzles in each unit, the total number of small-bore nozzles in all three units should be 81. It is not clear whether the exact number of small-bore nozzles on the hot leg piping is 63 or 81. (a) Provide the exact number of small-bore nozzles in the hot leg piping in each unit, the number of small-bore nozzles that have been repaired in each unit, and the number of small-bore nozzles that have not been repaired. (b) Discuss whether any small-bore nozzles in hot leg piping that are not covered under Relief Request 31. Confirm that the small bore nozzles in hot leg piping that are not covered under Relief Request 31, were analyzed for TLAA.

Appendix A---Updated Final Safety Analysis

The staff reviewed Appendix A3 of the License Renewal Application and did not find any discussion regarding leak-before-break (LBB) evaluations of the RCS piping as is discussed in Section 4.3.2.15. Justify why the TLAA of the LBB evaluation of the RCS piping is not included in Section A3 of Appendix A of the License Renewal Application.

cc:

Mr. Steve Olea
Arizona Corporation Commission
1200 W. Washington Street
Phoenix, AZ 85007

Mr. Douglas Kent Porter, Senior Counsel
Southern California Edison Company
Law Department, Generation Resources
P.O. Box 800
Rosemead, CA 91770

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 40
Buckeye, AZ 85326

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
612 E. Lamar Blvd., Suite 400
Arlington, TX 76011-4125

Chairman
Maricopa County Board of Supervisors
301 W. Jefferson, 10th Floor
Phoenix, AZ 85003

Mr. Aubrey V. Godwin, Director
Arizona Radiation Regulatory Agency
4814 S. 40th Street
Phoenix, AZ 85040

Mr. Scott Bauer, Director
Regulatory Affairs
Palo Verde Nuclear Generating Station
Mail Station 7636
P.O. Box 52034
Phoenix, AZ 85072-2034

Mr. Dwight C. Mims, Vice President
Regulatory Affairs and Plant Improvement
Palo Verde Nuclear Generating Station
Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

Mr. John C. Taylor, Director, Nuclear
Generation
El Paso Electric Company
340 E. Palm Lane, Suite 310
Phoenix, AZ 85004

Mr. James Ray
Public Service Company of New Mexico
2401 Aztec NE, MS Z110
Albuquerque, NM 87107-4224

Mr. Geoffrey M. Cook
Southern California Edison Company
5000 Pacific Coast Highway, Bldg. D21
San Clemente, CA 92672

Mr. Robert Henry
Salt River Project
6504 E. Thomas Road
Scottsdale, AZ 85251

Mr. Jeffrey T. Weikert
Assistant General Counsel
El Paso Electric Company
Mail Location 167
123 W. Mills
El Paso, TX 79901

Mr. Eric Tharp
Los Angeles Department of Water & Power
Southern California Public Power Authority
P.O. Box 51111, Room 1255-C
Los Angeles, CA 90051-0100

Mr. Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 N. Congress Avenue
Austin, TX 78701-3326