June 2, 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, AND 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 is reviewing the information contained in the license renewal application and has identified in the enclosure area where additional information is needed to complete the review. Further requests for additional information may be issued in the future.

A mutually agreeable date for your response, as discussed with Angela Krainik of APS staff, was determined to be 30 calendar days from the date of this letter. If you have any questions, please contact me at 301-415-1906 or by e-mail at Lisa.Regner@nrc.gov.

Sincerely,

/RA/

Lisa M. Regner, Sr. 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

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DATE	05/18/10	05/24/10	05/26/10	06/02/10	06/02/10

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Letter to R. Edington from L. Regner dated June 2, 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, AND ME0256)

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

PALO VERDE NUCLEAR GENERATING STATION (PVNGS)
LICENSE RENEWAL APPLICATION (LRA)
REQUEST FOR ADDITIONAL INFORMATION (RAI)

RAI 4.3-1

Issue
In the public meeting between Arizona Public Service Company (APS) and the U.S. Nuclear Regulatory Commission (NRC) held on Thursday, May 6, 2010, APS indicated that it had updated the design basis transients for the metal fatigue time-limited aging analysis (TLAA) to be consistent with those listed in the updated final safety analysis report (UFSAR) for the facility. Further, APS stated that the updated transient projection basis is based on the applicant's updated transient recount activities for the TLAA. The applicant clarified that the 25 percent assumed transient occurrence basis used in the original TLAA was only applied to five or six transients for which recount data could not be found.

Request
Clarify which of the transients in Tables 4.3-2 and 4.3-3 of the LRA (as modified by Amendment 14) the 25 percent assumed transient occurrence basis remains applicable to and justify why the application of this assumption is considered to yield a conservative 60-year cycle occurrence basis for these transients.

RAI 4.3-2

Issue
Table 4.3-3 in Amendment 14 of the LRA provides an adequate technical basis that PVNGS operates as a base load plant and that Transient No. 3, "5 percent per minute power ramp increase, from 15 percent to 100 percent power," and Transient No. 4, "5 percent per minute power ramp decrease, from 15 percent to 100 percent power," do not need to be counted relative to the 15,000 cycle limits for these transients. However, it appears that technical specification (TS) 5.5.5 and UFSAR Section 3.9.1.1 may still require these transients to be counted, specifically because these transients are currently listed as transients in Section I and II of UFSAR Table 3.9-1.

Section 4.3.2.1 of the LRA states that for the Unit 1 instrument nozzles, the calculated cumulative usage factor (CUF) of 0.68 is based on this 15,000 load following cycle limit. However, there is a factor of five difference in the CUF that is reported for these components for Unit 1 and those that are reported for the instrument nozzles at Units 2 and 3.

Request
1. Clarify, with justification, whether these transients are required to be counted per TS 5.5.5 and UFSAR Section 3.9.1.1. If these transients are required to be counted per TS 5.5.5 and UFSAR Section 3.9.1.1, clarify the actions that will be taken to resolve the inconsistency if it is determined there is a valid technical basis for not counting these transients.

ENCLOSURE

2. Clarify whether either Transient No. 3 or Transient No. 4 has occurred at the PVNGS site to date. If either transient has occurred, clarify how this is consistent with the plant being operated as a base load plant and justify not counting these transients.

3. Clarify why there is a factor of five difference between the CUFs reported for the instrument nozzles at Unit 1 from those that are reported for the corresponding nozzles at Units 2 and 3.

RAI 4.3-3

Issue
Section 4.3.5 of the LRA states that the calculated stresses in limiting locations were less than allowable in the revised design analyses for the reactor coolant hot leg sample lines piping and the steam generator (SG) downcomer and feedwater recirculation lines piping. However, LRA Section 4.3.5 does not provide sufficient information for the staff to confirm these assertions.

Request
Provide the code allowable stress limits and the stress ranges obtained in the revised design analyses for the reactor coolant hot leg sample line piping and the SG downcomer and feedwater recirculation line piping. Also, provide the American Society of Mechanical Engineer Code edition and specific subsection used for the revised design analyses for these piping components.

RAI 4.3-4

Issue
Section 4.3.4 of the LRA states that for reactor pressure vessel (RPV) shell and lower head, RPV inlet and outlet nozzles, and safety injection nozzle (forging knuckle), the maximum applicable environmental factors (F_{en}) for low alloy steel was used and was determined following NUREG/CR-6583, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels.” However, LRA Section 4.3.4 does not provide sufficient information to confirm this statement.

Request
Demonstrate that the F_{en} factor used for assessment of the reactor coolant environment impact on the RPV shell and lower head, RPV inlet and outlet nozzles, and safety injection nozzle (forging knuckle) are the maximum applicable for a given material. Provide a basis and justification for any assumptions that were made for the parameters in the assessment, such as strain rate, dissolved oxygen, temperature and sulfur content.

RAI 4.3-5

Issue
Note 7 and 9 of Table 4.3-11 of the LRA provides the reanalysis computed F_{en} values for load set pairs with a significant fatigue contribution for the charging system nozzle (safe end) and the safety injection nozzle (safe end), respectively. Section 4.3.4 of the LRA does not contain sufficient information on the assumptions that have been used for the environmental F_{en} factor calculations.

Request

1. Describe in detail the methodology that has been used for the environmental F_{en} factor calculation of the charging system nozzle and the safety injection nozzle.

2. Provide a basis for any assumptions that were made for the parameters, such as strain rate, dissolved oxygen, and temperature, in the assessment of a computed F_{en} value for the load set pairs with a significant fatigue contribution.

3. Confirm the value of the maximum F_{en} factor used for all remaining load set pairs.

RAI 4.3-6

Background

LRA Section 4.3.4 states that a bounding F_{en} factor of 1.49 was used for the Alloy 600 component, pressurizer heater penetrations. NUREG/CR-6335, "Fatigue Strain–Life Behavior of Carbon and Low–Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," provides the statistical characterizations used to derive this F_{en} factor of 1.49 for Alloy 600, and states the fatigue S-N database (fatigue per load cycle curves) for Alloy 600 is extremely limited and does not cover an adequate range of material and loading variables that might influence fatigue life. It further states that the data was obtained from relatively few heats of material and are inadequate to establish the effect of strain rate on fatigue life in air or of temperature in a water environment. NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," incorporates more recent fatigue data using a larger database for determining the F_{en} factor of nickel alloys.

Issue

The F_{en} factor of 1.49 for nickel alloys may be non-conservative. The F_{en} for nickel alloys based on NUREG/CR-6909 varies based on temperature, strain rate and dissolved oxygen. Based on actual plant operating conditions the F_{en} factor can vary from a value of 1.0 to 4.52 based on this methodology. Therefore, the CUF value for the pressurizer heater penetrations may be as high as 2.86 using the CUF presented in the LRA and the maximum F_{en} derived from NUREG/CR-6909 which would exceed the design limit of 1.0 when considering environmental effects of reactor coolant during the period of extended operation.

Request

1. Since the F_{en} for nickel alloys can vary from 1.0 to 4.52 based on NUREG/CR-6909 and the CUF value may exceed the design limit of 1.0 for the pressurizer heater penetrations when considering environmental effects, justify using a value of 1.49 for the F_{en} factor for this nickel alloy component.

2. Describe the current or future planned actions to update the CUF calculation with F_{en} factor for the Alloy 600 component only, consistent with the methodology in NUREG/CR-6909. If there are no current or future planned actions to update the CUF calculation with F_{en} factor for the Alloy 600 component consistent with the methodology in NUREG/CR-6909, provide a justification for not performing the update.

cc:

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