September 11, 2007

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

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - CORRECTION TO THE SAFETY EVALUATIONS ASSOCIATED WITH ISSUANCE OF AMENDMENT NOS. 149 AND 157 RE: UPRATED POWER OPERATIONS (TAC NOS. MD4491, MD4492, AND MD4493)

Dear Mr. Edington:

On September 29, 2003, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 149 to Facility Operating License No. NPF-51 for the Palo Verde Nuclear Generating Station (Palo Verde), Unit 2. Subsequently, on November 16, 2005, the Commission issued Amendment No. 157 to Facility Operating License No. NPF-41, Amendment No. 157 Facility Operating License No. NPF-51, and Amendment No. 157 to Facility Operating License No. NPF-74, for Palo Verde, Units 1, 2, and 3, respectively. The amendments consisted of changes to the facility operating licenses and Technical Specifications (TSs) to support replacement of steam generators and subsequent operation at an increased power level of 3990 megawatts thermal (MWt), a 2.94 percent increase from the 3876 MWt for Palo Verde, Units 1, 2, and 3. Amendment No. 157 also made administrative changes to the TSs for Palo Verde, Unit 2 so that the changed pages would apply to all three Palo Verde units.

In correspondence dated January 26 and February 21, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML072480541 and ML0724480553, respectively), Arizona Public Service Company notified NRC of an inaccuracy in the safety evaluations (SEs) for Amendment Nos. 149 and 157. This inaccuracy involved the addition of a sentence by the NRC staff, in its safety evaluations, that was inconsistent with the licensee's submittals with regard to the limiting anticipated operational occurrence (AOO) for a single failure event and the maximum linear heat rate produced by a control element assembly withdrawal (CEAW). This matter was discussed with Mr. Glenn Michael of your staff on March 14, 2007.

The NRC staff agrees that deletion of the subject sentence in the staff's safety evaluations sets the record straight by removing a statement that was not part of the licensee's submittal. Correcting the description of the licensee's analysis does not impact the NRC staff's conclusion that the appropriate regulatory requirements were met. This correction does not change the NRC staff's conclusions regarding Amendment Nos. 149 and 157. The NRC staff agrees that the deletion of this phrase is appropriate and is reissuing the affected pages of the SEs. On page 53 of Amendment No. 149, the sentence, "This is combined with the maximum linear heat rate produced by a CEAW," is deleted. On page 57 of Amendment No. 157, the sentence, "This is combined with the maximum linear heat rate (LHR) produced by a CEAW," is deleted. Enclosed is a corrected version of page 53 of the SE for Amendment No. 149 and page 57 of

the SE for Amendment No. 157, with the correction denoted by a vertical bar. Please discard the associated page from the previous SEs and replace it with the enclosed pages.

If you have any questions, please call me at (301) 415-5723.

Sincerely,

/RA/

Michael T. Markley, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures: As stated

cc w/encls: See next page

the SE for Amendment No. 157, with the correction denoted by a vertical bar. Please discard the associated page from the previous SEs and replace it with the enclosed pages.

If you have any questions, please call me at (301) 415-5723.

Sincerely,

/RA/

Michael T. Markley, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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and STN 50-530

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ADAMS Accession No.: Pkg ML072480376 (Ltr ML072580360

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/LPL4/BC
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DATE	9/4/07	9/11/07	9/11/07

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RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded, (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to ensure that the SAFDLs are not exceeded, (4) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (6) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

The licensee created a composite limiting transient to bound the MDNBR of infrequent events (including AOOs in combination with a single failure). It is assumed that the unspecified event degrades the DNBR to the SAFDL level. The most limiting single failure is LOP, resulting in the coast-down of all RCPs.~~ This is combined with the maximum linear heat rate produced by a CEAW. ~~No single failures or operator errors can degrade DNBR more than the above circumstances; therefore, no other failures are assumed. Initial conditions conservatively assume a 116 percent power level due to a preexisting condition from the undefined AOO and a turbine trip coincident with reactor trip, although a 3 second delay exists. No operator action is assumed for 30 minutes after transient initiation.

The acceptance criteria are those for infrequent events (including AOOs with single failure), i.e., limited fuel damage and maximum RCS pressure within 110 percent of the RCS design value.

The analysis is based on the CENTS code supplemented by CETOP-D for DNBR (using the CE-1 CHF correlation), and the HERMITE code for the calculation of the initial conditions. The MDNBR is calculated using the more detailed TORC code.

Because such events are heat-up transients, it is implicitly postulated that the PSVs will keep the maximum pressure within acceptable limits. The results are comparable to those for a broken RCP shaft, i.e., limited fuel damage and no MDNBR propagation are predicted. The maximum pressure is within acceptance limits because the PSVs have sufficient capacity to relieve overpressure. These transient analyses provides confidence that the limiting infrequent events (including AOOs in combination with a single failure) are well within prescribed limits.

The NRC staff has reviewed the licensee's analyses of the limiting hypothetical AOO transient with LOP. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system will continue to ensure that the MDNBR and the peak RCS pressure will remain within the acceptance limits for this hypothetical event. In addition, core geometry and long-term cooling will remain within acceptable limits for such an event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 26, 27, and 31 for this hypothetical event. Therefore, the NRC staff finds the proposed bounding transient acceptable.

4.3.7 Limiting Infrequent Events

4.3.7.1 AOOs in Combination With a Single Active Failure

The limiting infrequent event is designed to test the plant's capability to respond to extreme transient conditions. The acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded, (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to ensure that the SAFDLs are not exceeded, (4) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (6) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

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The acceptance criteria are those for infrequent events (including AOOs with single failure), i.e., limited fuel damage and maximum RCS pressure within 110 percent of the RCS design value.

The analysis is based on the CENTS code supplemented by CETOP-D for DNBR (using the CE-1 CHF correlation), and the HERMITE code for the calculation of the initial conditions. The MDNBR is calculated using the more detailed TORC code.

Because such events are heat-up transients, it is implicitly postulated that the PSVs will keep the maximum pressure within acceptable limits. The results are comparable to those for a broken RCP shaft, i.e., limited fuel damage and no MDNBR propagation are predicted. The maximum pressure is within acceptance limits because the PSVs have sufficient capacity to relieve overpressure. These transient analyses provides confidence that the limiting infrequent events (including AOOs in combination with a single failure) are well within prescribed limits.

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