APS

Palo Verde Nuclear Generating Station

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10 CFR 50.55a(a)(3)(i)

102-05503-CDM/SAB/RJR
May 26, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

References:

1. APS letter 102-05486-CDM/SAB/RJR, Request to Extend the Second 10-Year, American Society of Mechanical Engineers Section XI, Inservice Inspection Program Interval for Reactor Vessel Weld Examinations – Relief Request No. 34, dated May 4, 2006.
2. Letter from Nuclear Regulatory Commission to Westinghouse Electric Company, "Summary of teleconference with the Westinghouse Owners Group regarding potential one cycle relief of reactor pressure vessel shell weld inspections at pressurized water reactors related to WCAP-16168-NP, 'Risk-Informed Extension of Reactor Vessel In-Service Inspection Intervals," dated January 27, 2005.

Dear Sirs:

SUBJECT: Palo Verde Nuclear Generating Station (PVNGS)
Units 2 and 3
Docket Nos. STN 50-529/530
Additional Information to Support the Request to Extend the Second 10-Year, American Society of Mechanical Engineers Section XI, Inservice Inspection Program Interval for Reactor Vessel Weld Examinations – Relief Request No. 34

As a result of conversations with the NRC staff, Arizona Public Service Company (APS) is providing additional information in support of its May 4, 2006 request (Reference 1) to use an alternative to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Paragraph IWB-2412, Inspection Program B, for PVNGS Units 2 and 3.

In Reference 2 that the staff agreed to licensees submitting relief requests for a one cycle extension of the reactor vessel weld examinations to provide additional time for completing evaluations and staff review associated with Reference 1, the Westinghouse Owners Group Topical Report, WCAP - 16168.

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APS' technical justification to extend PVNGS's second inspection interval performance of Category B-A and B-D examinations by one fuel cycle provided in Reference 1 was not complete in that the additional information requested by the NRC from other licensees was not contained in APS' original request. This additional information is being provided in the enclosure to this letter and addresses the accident sequences identified by the NRC during its re-evaluation of the risk from pressurized thermal shock (PTS).

This letter contains no new commitments and no revisions to existing commitments. APS is also changing the date by which approval of this request is needed to October 1, 2006, to support activities for the fall 2006 2R13 refueling outage.

If you have any questions about this change, please telephone Thomas N. Weber at (623) 393-5764.

Sincerely,

David Mauldin

CDM/SAB/RJR/gt

Enclosure: Relief Request No. 34 – Additional Information Addressing the Accident Sequences Identified by the NRC during its Re-evaluation of the Risk from Pressurized Thermal Shock (PTS).

cc:	B. S. Mallett	NRC Region IV Regional Administrator
	M. B. Fields	NRC NRR Project Manager
	G. G. Warnick	NRC Senior Resident Inspector for PVNGS

ENCLOSURE

Relief Request No. 34 – Additional Information Addressing the Accident Sequences Identified by the NRC during its Re-evaluation of the Risk from Pressurized Thermal Shock (PTS)

NRC QUESTION:

You stated that the technical justification for your request was consistent with the guidance provided in a January 27, 2005, letter from the NRC to Westinghouse Electric Company (Summary of Teleconference with the Westinghouse Owners Group Regarding Potential One Cycle Relief of Reactor Pressure Vessel Shell Weld Inspections at Pressurized-Water Reactors Related to WCAP-16168-NP, "Risk Informed Extension of Reactor Vessel In-Service Inspection Intervals"). Item number six of this guidance is repeated below:

> The licensee could then provide a discussion of how, based on its plant operational experience, fleet-wide operational experience, and plant characteristics, the likelihood of an event (in particular, a significant pressurized thermal shock event) over the next operating cycle which could challenge the integrity of the reactor vessel pressure vessel (RPV), if a flaw was present, is very low.

Section 5.5 of your submittal includes general statements indicating that the likelihood of pressurized thermal shock (PTS) events is small and briefly describes APS operating procedures that provide actions to avoid, or limit thermal shock to the reactor pressure vessel.

The NRC staff is re-evaluating the risk from PTS events in a study done to develop a technical basis for revising Title 10 of the *Code of Federal Regulations,* Part 50, Section 61 (10 CFR 50.61). Although the NRC staff has not yet completed its evaluation, the current results indicate that the following three types of accident sequences cause the more severe PTS events and thereby dominate the risk. Please describe the characteristics of your plant (design and operating procedures) that provide assurance that the likelihood of a severe PTS event over the next operating cycle which could challenge the integrity of the RPV, if a flaw was present, is very low.

Sequence 1:

> Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after about 1 hour. Severe PTS events also require the failure to properly control high-head injection.

Sequence 2:

> Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves. Severe PTS events also require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators) and failure to properly control high pressure injection.

Sequence 3:

Four to nine-inch loss-of-coolant accidents. Severity of PTS event depends on break location (worst location appears to be in the pressurizer surge line) and primary injection systems flow rate and water temperature.

APS Response:

As discussed in Section 5.5 of APS' May 4, 2006 request, PVNGS Units 2 and 3 have implemented emergency operating procedures (EOP) and operator training to prevent the occurrence of PTS events. Consistent with the Combustion Engineering (CE) Emergency Response Guidelines (ERGs), the PVNGS EOPs allow operators to identify the onset of PTS conditions and provide the steps required to mitigate any cold pressurization challenge to RV integrity. The basic PTS mitigation strategy of the PVNGS EOPs involves 1) termination of the primary system cool down, 2) termination of emergency core cooling system flow (if proper criteria are met), 3) depressurization of the primary system, 4) establishment of stable primary system conditions in the normal operating range, and 5) implementation of a thermal "soaking" period prior to any cool down outside of the normal operating region.

Sequence 1:

Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after about 1 hour. Severe PTS events also require the failure to properly control high-head injection.

Initially, the control room personnel complete procedure 40EP-9EO01, "Standard Post Trip Actions," (SPTAs). This procedure is used for any event which actuates or requires a reactor trip. It is intended that the operator check each Safety Function and perform the Contingency Actions if necessary. The crew would then enter 40EP-9EO03, "Loss of Coolant Accident," (LOCA). The goals of this procedure are to mitigate the effects of a LOCA, to isolate the break (if possible), and to establish either long term cooling using the safety injection system or the shutdown cooling system. After some verification and notification steps, the crew reaches Step 23 within a few minutes. Steps 23 through 25 include the following guidance:

23. Perform the following:

a. PERFORM Appendix 5, RCS and PZR Cooldown Log.	
b. Cooldown the Steam Generators using the SBCS.	b.1 Cooldown the Steam Generators using the ADVs by **ONE** of the following: • Operation from the Control Room • Appendix 18, Local ADV Operation

24. **IF** steaming to atmosphere,
    THEN inform Radiation Protection and
    the RMS Technician.

25. Depressurize the RCS to less than
    385 psia [385 psia] by performing the
    following:

    a. Operate Main or Auxiliary
       Pressurizer spray and PERFORM
       Appendix 6, Spray Valve
       Actuation Data Sheet.

    b. **IF** Safety Injection throttle criteria
       are met, **THEN** control **ANY** of the following
       to lower RCS pressure.
       • Charging and letdown flow
       • HPSI flow

Additionally, Step 27 includes the following guidance:

27. **IF** at least one HPSI Pump is operating,
    AND ALL of the following conditions
    exist:
    • RCS is 24°F or more subcooled
    • Pressurizer level is greater than
      10% and **NOT** lowering
    • At least one Steam Generator is
      available for RCS heat removal
      with level being maintained within
      or being restored to 45 - 60% NR
    • RVLMS indicates RVUH level is
      16% or more

    THEN throttle HPSI flow or stop the
    HPSI Pumps one pump at a time.

The steps listed above initiate an RCS cool down and depressurization and allows throttling or stopping high pressure safety injection (HPSI) flow as needed. Flow requirements to maintain the core covered and cooled will decrease as RCS pressure is lowered, or the pressurizer safety relief valve reseats during the cool down. Throttling and/or stopping HPSI prevents or minimizes the magnitude of re-pressurization of the RCS, thereby precluding PTS.

Sequence 2:

> Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves. Severe PTS events also require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators) and failure to properly control high pressure injection.

Initially, the control room personnel complete procedure 40EP-9EO01, "Standard Post Trip Actions." The Operating staff would then enter 40EP-9EO05, "Excess Steam Demand," (ESD). The goals of this procedure are to mitigate the effects of an ESD, maintain the plant in hot standby, or hot shutdown (if the break has been isolated), or to establish Shutdown Cooling System entry conditions while minimizing radiological releases to the environment and maintaining adequate core cooling. 40EP-9EO05 includes the following guidance in Step 14 which is performed after isolating the most affected Steam Generator, including stopping auxiliary feedwater to the faulted SG:

14. Stabilize RCS temperature using the lowest Tc by performing the following:

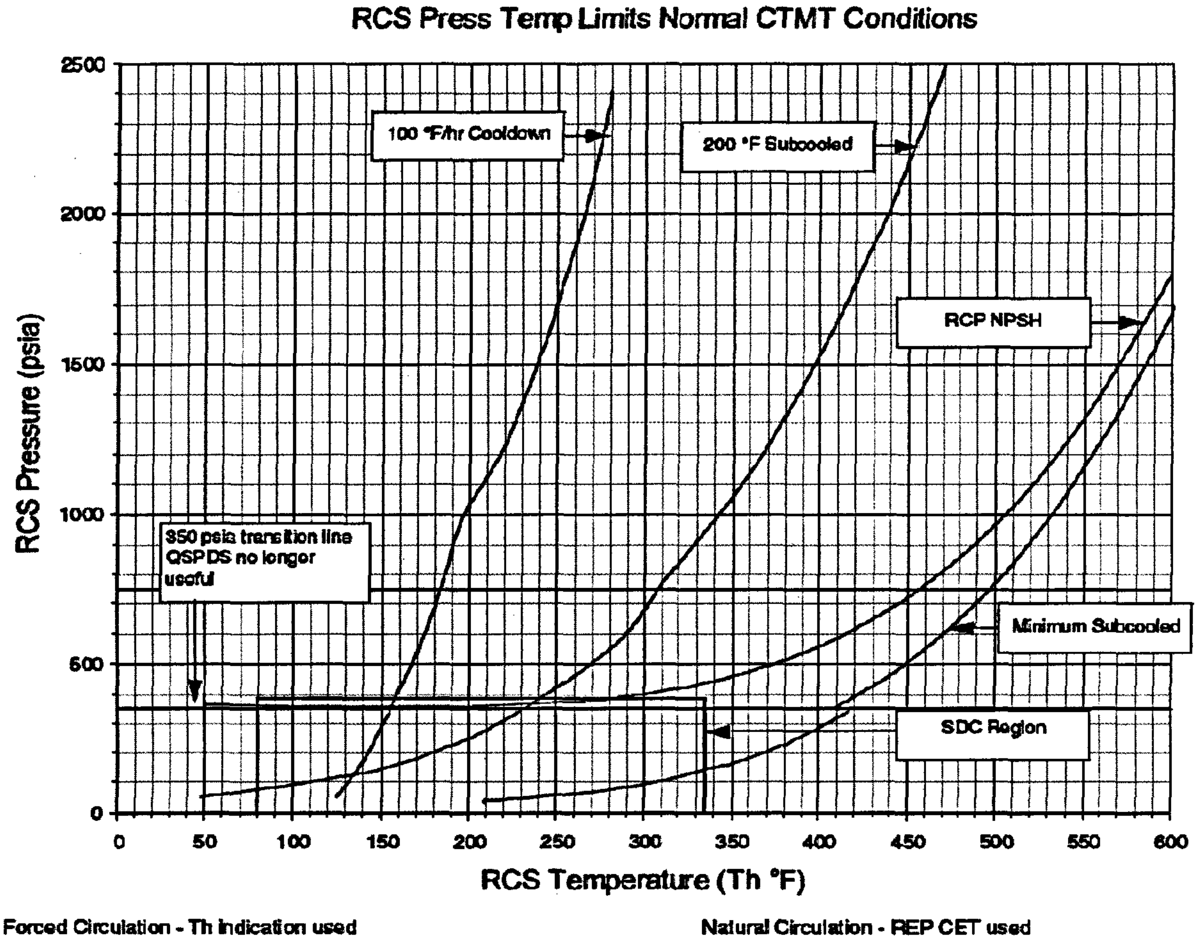
    a. Maintain Tc within the P/T limits. REFER TO Appendix 2, Figures

    b. Steam the least affected Steam Generator using **ANY** of the following:

        - SBCS
        - ADVs from the Control Room
        - Appendix 18, Local ADV Operation

Stabilizing Tcold within the Pressure/Temperature (P/T) limits precludes PTS. EOP 40EP-9EO05 also contains the guidance for throttling HPSI when throttle criteria are met (same as Step 27 from LOCA). EOP 40EP-9EO10, "Standard Appendices," contains all of the figures, tables, charts, graphs and sub-procedures associated with performance of the EOPs.

Appendix 2 of 40EP-9EO10 shows the acceptable areas of operation and delineates where PTS becomes a concern (the 200 degree subcooled line).
Appendix 2 is provided below:



Sequence 3:

Four to nine-inch loss-of-coolant accidents. Severity of PTS event depends on break location (worst location appears to be in the pressurizer surge line) and primary injection systems flow rate and water temperature.

The response to this sequence would be the same as Sequence 1.

Additionally, as part of a recent power uprate (PUR) amendment request, APS performed fluence calculations using the existing analysis of record (AOR) at the 4200 MWt power level. An out-in type of fuel loading was assumed, however, the proposed PUR was for 3990 MWt and the loading pattern has been low leakage for a number of cycles. Both conditions are conservative and the AOR bounds the values calculated for the PUR. With the issuance of Amendment No. 157 to Facility Operating License Nos. NPF-51, and NPF-74, dated November 16, 2005, the NRC acknowledged that the P-T curves currently approved for Units 2 and 3 were valid for 32 effective full power years (EFPY). Since Units 2 and 3 are estimated to be at only 17.3 and 17.6 EFPY respectively at the end of operating cycle 14, sufficient margin exists until the examinations are performed during 14^{th} refueling outages for each unit.