UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-51

AND AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By letter dated March 13, 1990, the Arizona Public Service Company (APS or the licensee) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), requested changes to the Technical Specifications for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (Appendix A to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, respectively). The proposed changes would update the Reactor Vessel Pressure-Temperature (P/T) curves and Low Temperature Overpressure Protection (LTOP) enable temperatures, in accordance with the irradiation damage prediction methodology of Revision 2 of Regulatory Guide 1.99 "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

The Reactor Coolant System (RCS) pressure-temperature limits during plant heatup and cooldown are specified in Technical Specification 3.4.8.1 for the Palo Verde Units. The pressure-temperature curves in the current Technical Specifications are based on an assumed design basis neutron fluence through 10 effective full power years (EFPY). The proposed amendments change the effectiveness of the P/T limits of 8 and 32 effective full power years (EFPY). The licensee proposed to use one set of P/T limits for all three units. The proposed P/T limits were developed based on Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

APS provided its updated pressure-temperature curves in proposed Technical Specification Figure 3.4-2a (for less than 8 EFPY) and Figure 3.4-2b (for 8 to 32 EFPY), changes in the values of the RCS cold leg temperature at which LTOP should be enabled, and the justification for the changes. New heatup and cooldown rates as a function of indicated reactor coolant temperature are also proposed in an updated Table 3.4-3.

2.0 EVALUATION OF THE P/T LIMITS

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in PVNGS 1, 2, and 3 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the limiting materials at 8 EFPY and 32 EFPY for all three units were Unit 1 intermediate shell plates M-6701-1 with 0.07% copper (Cu), 0.66% nickel (Ni), and an initial RT_{ndt} of 30°F; and plate M-6701-2 with 0.06% Cu, 0.61% Ni, and an initial RT_{ndt} of 40°F.

The licensee has not removed any surveillance capsules from PVNGS 1, 2, and 3 because none of the units has reached the removal date in the capsule withdrawal schedule. The staff has ascertained that all surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

At 8 EFPY, the staff calculated the highest ART to be 96.8°F and 72.2°F at the 1/4T (T= reactor vessel beltline thickness) and at 3/4T locations, respectively. At 32 EFPY, the staff calculated the ART to be 116°F and 97.8°F at the 1/4T and 3/4T locations. The staff used a neutron fluence of 4.2E18 n/cm^2 at 1/4T and 1.09E18 n/cm^2 at 3/4T at 8 EFPY. The staff used a neutron fluence of 1.68E19 n/cm^2 at 1/4T and 4.38E18 at 3/4T at 32 EFPY. The ART was determined by Section 1 of RG 1.99, Rev. 2 because no surveillance capsules have been removed.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 102°F and 90°F for the 1/4T and 3/4T locations at 8 EFPY, and 116°F and 103°F for the 1/4T and 3/4T locations at 32 EFPY. The licensee identified plates M-6701-2 and M-6701-3 as the limiting materials. The difference between the staff and licensee's limiting materials selection and ARTs is because the licensee used a different safety margin in calculating ART. The staff considers the licensee's limiting materials and ARTs acceptable. Substituting the licensee's ARTs into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the highest flange reference temperature of -10°F for Unit 3, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Using the method in RG 1.99, Rev. 2, the lowest predicted Charpy USE of all beltline materials is plate M-6701-1 from Unit 1 with 62.2 ft-lb. This is above 50 ft-lb and, therefore, is acceptable.

3.0 EVALUATION OF LTOP

LTOP is provided by relief valves on the Shutdown Cooling System (SCS) lines. These relief valves are set at a pressure low enough to prevent violation of the Appendix G heatup and cooldown curves should a RCS pressure transient occur during low temperature operations. The licensee, in its March 13, 1990 submittal, identified the most limiting overpressure

[illegible]

transients analyzed to determine the relief valve setpoint for LTOP. The relief valve setpoint limit has been previously set by analysis of the limiting transients for mass addition and energy addition. Technical Specification 3.4.8.3 currently requires that two relief valves shall be OPERABLE with the setpoint selected for the low temperature mode of operation. The modified Technical Specification 3.4.8.3 maintains the same pressure setpoint and revises the values of the applicable temperatures for LTOP based on a reanalysis of the limiting transients.

The most limiting mass addition transient was analyzed assuming two High Pressure Safety Injection (HPSI) pumps injecting into a water solid RCS with full charging capacity and with the letdown isolated. The transient analysis is typically performed to determine the pressure overshoot past the LTOP setpoint such that the Appendix G curves are not exceeded during the transient.

The energy input transient was analyzed assuming a 100°F temperature difference between the steam generator and the RCS cold leg. A reactor coolant pump startup in one loop was assumed in order to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot is calculated such that the Appendix G pressure-temperature curves for each Unit are not exceeded.

The licensee's analyses were performed using the same methodology as the prior application for ten EFPY. For the revised analyses the LTOP enable temperatures were determined by following the guidance that for LTOP, the enable temperature is the water temperature corresponding to a metal temperature at the vessel beltline that is controlling in the Appendix G calculations. The resulting enable temperatures were calculated by the licensee to be 291°F during heatup and 214°F during cooldown. The results indicated that a change in the present SCS relief valve setpoint of 467 psig is not required. A footnote is added to Technical Specifications 3.4.1.3 and 3.4.1.4.1 which states:

> Reactor Coolant Pump operation is limited to 2 Reactor Coolant Pumps with RCS cold leg temperature less than or equal to 200°F, 3 Reactor Coolant Pumps with RCS cold leg temperature greater than 200°F but less than or equal to 500°F.

This note is added to maintain the analysis assumptions of the flow induced pressure correction factors due to Reactor Coolant Pump operation.

The licensee-proposed changes in Technical Specifications 3.4.1.3, 3.4.1.4.1, 3.4.8.1, 3.4.8.3 and 4.4.8.3 and the associated bases sections reflect the above discussed LTOP alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.4-2a and 3.4-2b, and Table 3.4-3 in Technical Specification 3.4.8.1. The staff finds that they are reasonably conservative and acceptable.

[illegible]

4.0 CONCLUSION

Based on the staff evaluation in Section 2.0, the staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 8 EFPY and 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the PVNGS 1, 2, and 3 Technical Specifications.

Based on the staff evaluation in Section 3.0, the staff concludes that the proposed Technical Specifications 3.4.1.3, 3.4.1.4.1, 3.4.8.1, 3.4.8.3, and 4.4.8.3.1 and their associated bases are acceptable to support the updated pressure-temperature limits identified in Technical Specification Figures 3.4-2a and 3.4-2b applicable for a period up to 32 EFPY.

5.0 CONTACT WITH STATE OFFICIAL

The Arizona Radiation Regulatory Agency has been advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments involve changes in a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amount, and no significant change in the type, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendments involves no significant hazard consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

8.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits

3. Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report

4. January 31, 1989, Letter from D. B. Karner (APS) to USNRC Document Control Desk, Subject: Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Generic Letter 88-11

5. March 13, 1990, Letter from W. F. Conway (APS) to USNRC Document Control Desk, Subject: Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Proposed Technical Specification to Incorporate the Requirements of Generic Letter 88-11

Principal Contributors: M. McCoy
J. Tsao

Dated: July 25, 1990