UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 25, 2010

Mr. Randall K. Edington
Executive Vice President Nuclear/
Chief Nuclear Officer
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 - ISSUANCE OF AMENDMENTS RE: RELOCATION OF CERTAIN TECHNICAL SPECIFICATION REQUIREMENTS TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (TAC NOS. ME0698, ME0699, AND ME0700)

Dear Mr. Edington:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 178 to Facility Operating License No. NPF-41, Amendment No. 178 to Facility Operating License No. NPF-51, and Amendment No. 178 to Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 19, 2009, as supplemented by letters dated December 22, 2009, and February 23, 2010.

The amendments revise the TSs to relocate the reactor coolant system pressure and temperature (P/T) limits and the low temperature overpressure protection (LTOP) enable temperatures to a licensee-controlled document outside of the TSs. The P/T limits and LTOP enable temperatures will be specified in a Pressure and Temperature Limits Report (PTLR) that will be located in the PVNGS Technical Requirements Manual and administratively controlled by a new TS 5.6.9, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." The proposed changes are in accordance with the guidance in NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996.

In your application of February 19, 2009, you also submitted a related request for exemption from certain requirements of Title 10 of the *Code of Federal Regulations*, Part 50, Appendix G, "Fracture Toughness Requirements." That exemption request will be addressed by the NRC staff in separate correspondence.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

James R. Hall

James R. Hall, 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:
1. Amendment No. 178 to NPF-41
2. Amendment No. 178 to NPF-51
3. Amendment No. 178 to NPF-74
4. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. NPF-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated February 19, 2009, as supplemented by letters dated December 22, 2009, and February 23, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating License No. NPF-41 and Technical Specifications

Date of Issuance: February 25, 2010

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. NPF-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated February 19, 2009, as supplemented by letters dated December 22, 2009, and February 23, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C(2) of Facility Operating License No. NPF-51 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating License No. NPF-51 and Technical Specifications

Date of Issuance: February 25, 2010

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. NPF-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated February 19, 2009, as supplemented by letters dated December 22, 2009, and February 23, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 3

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C(2) of Facility Operating License No. NPF-74 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 150 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating
License No. NPF-74 and
Technical Specifications

Date of Issuance: February 25, 2010

ATTACHMENT TO LICENSE AMENDMENT NOS. 178, 178, AND 178

FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

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

Replace the following pages of the Facility Operating Licenses Nos. NPF-41, NPF-51, and NPF-74, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Operating Licenses

REMOVE	INSERT

Replace Page 5 of Facility Operating License No. NPF-41 with the attached Page 5.

Replace Page 6 of Facility Operating License No. NPF-51 with the attached Page 6.

Replace Page 4 of Facility Operating License No. NPF-74 with the attached Page 4.

Technical Specifications

REMOVE	INSERT
1.1-6	1.1-6
3.4.3-1	3.4.3-1
3.4.3-2	3.4.3-2
3.4.3-3	---
3.4.3-4	---
3.4.3-5	---
3.4.3-6	---
3.4.3-7	---
3.4.6-1	3.4.6-1
3.4.7-1	3.4.7-1
3.4.11-1	3.4.11-1
3.4.13-1	3.4.13-1
5.6-8	5.6-8
---	5.6-9

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This license is subject to the antitrust conditions delineated in Appendix C to this license.

(4) Operating Staff Experience Requirements

Deleted

(5) Post-Fuel-Loading Initial Test Program (Section 14, SER and SSER 2)*

Deleted

(6) Environmental Qualification

Deleted

(7) Fire Protection Program

APS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as supplemented and amended, and as approved in the SER through Supplement 11, subject to the following provision:

APS may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Emergency Preparedness

Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This license is subject to the antitrust conditions delineated in Appendix C to this license.

(4) Operating Staff Experience Requirements (Section 13.1.2, SSER 9)*

Deleted

(5) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(6) Fire Protection Program

APS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as supplemented and amended, and as approved in the SER through Supplement 11, subject to the following provision:

APS may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(7) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER and SSER 9)

Deleted

(8) Supplement No. 1 to NUREG-0737 Requirements

Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3876 megawatts thermal (100% power) through operating cycle 13, and 3990 megawatts thermal (100% power) after operating cycle 13, in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This license is subject to the antitrust conditions delineated in Appendix C to this license.

(4) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 171, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

(6) Mitigation Strategy License Condition

APS shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance.
2. Assessment of mutual aid fire fighting assets.
3. Designated staging areas for equipment and materials.
4. Command and control.
5. Training of response personnel.

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the site specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.9.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt for Unit 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 3990 MWt for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full strength CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part length or part strength CEA position.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times; except when reactor vessel head is fully detensioned such that the RCS cannot be pressurized.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < 500 psia.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-----------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in the PTLR.	30 minutes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops – MODE 4

LCO 3.4.6 Two loops or trains consisting of any combination of RCS loops and shutdown cooling (SDC) trains shall be OPERABLE and at least one loop or train shall be in operation.

---------------------------NOTES---------------------------

1. All reactor coolant pumps (RCPs) and SDC pumps may be de-energized for ≤ 1 hour per 8 hour period, provided:
   a. No operations are permitted that would cause reduction of the RCS boron concentration; and
   b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. No RCP shall be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless the secondary side water temperature in each Steam Generator (SG) is < 100°F above each of the RCS cold leg temperatures.
3. No more than 2 RCPs may be in operation with RCS cold leg temperature ≤ 200°F. No more than 3 RCPs may be in operation with RCS cold leg temperature > 200°F but ≤ 500°F.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable. AND Two SDC trains inoperable.	A.1 Initiate action to restore a second loop or train to OPERABLE status.	Immediately

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops – MODE 5, Loops Filled

LCO 3.4.7 One Shutdown Cooling (SDC) train shall be OPERABLE and in operation, and either:

a. One additional SDC train shall be OPERABLE; or

b. The secondary side water level of each Steam Generator (SG) shall be ≥ 25%.

---------------------------NOTES---------------------------

1. The SDC pump of the train in operation may be de-energized for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required SDC train may be inoperable for up to 2 hours for surveillance testing provided that the other SDC train is OPERABLE and in operation.

3. No Reactor Coolant Pump (RCP) shall be started with one or more of the RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless the secondary side water temperature in each SG is < 100°F above each of the RCS cold leg temperatures.

4. No more than 2 RCPs may be in operation with RCS cold leg temperature ≤ 200°F. No more than 3 RCPs may be in operation with RCS cold leg temperature > 200°F but ≤ 500°F.

5. All SDC trains may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Safety Valves-MODE 4

LCO 3.4.11 One pressurizer safety valve shall be OPERABLE with a lift setting ≥ 2450.25 psia and ≤ 2549.25 psia.

APPLICABILITY: MODE 4 with all RCS cold leg temperatures greater than the LTOP enable temperature specified in the PTLR.

-----------------------------NOTE-----------------------------
The lift settings are not required to be within LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 72 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. All pressurizer safety valves inoperable.	A.1 Be in MODE 4 with one Shutdown Cooling System suction line relief valve in service.	Immediately
	<u>AND</u>	
	A.2 Perform SR 3.4.11.2 and SR 3.4.11.3 for the required Shutdown Cooling System suction line relief valve to comply with Action A.1.	Immediately
	<u>AND</u>	
	A.3 Be in MODE 4 with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.	8 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.13 An LTOP System shall be OPERABLE consisting of:

a. Two OPERABLE Shutdown Cooling System suction line relief valves with lift settings ≤ 467 psig aligned to provide overpressure protection for the RCS; or

b. The RCS depressurized and an RCS vent of ≥ 16 square inches.

---------------------------NOTE---------------------------
No RCP shall be started unless the secondary side water temperature in each steam generator (SG) is ≤ 100°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR.
MODE 5,
MODE 6 when the reactor vessel head is on.

---------------------------NOTE---------------------------
LCO 3.0.4.b is not applicable when entering MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required Shutdown Cooling System suction line relief valve inoperable in MODE 4.	A.1	Restore required Shutdown Cooling System suction line relief valve to OPERABLE status.	7 days

(continued)

5.6 Reporting Requirements (continued)

5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.10, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG.

b. Active degradation mechanisms found.

c. Nondestructive examination techniques utilized for each degradation mechanism.

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications.

e. Number of tubes plugged during the inspection outage for each active degradation mechanism.

f. Total number and percentage of tubes plugged to date.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

(continued)

5.6 Reporting Requirements (continued)

5.6.9 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following Technical Specifications (TSs):

1. TS 3.4.3, RCS Pressure and Temperature (P/T) Limits;
2. TS 3.4.6, RCS Loops - Mode 4;
3. TS 3.4.7, RCS Loops - Mode 5 Loops Filled;
4. TS 3.4.11, Pressurizer Safety Valves - Mode 4; and
5. TS 3.4.13, Low Temperature Overpressure Protection (LTOP) System.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

- CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. NPF-51, AND

AMENDMENT NO. 178 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 February 19, 2009 (Reference 1), as supplemented by letters dated December 22, 2009 (Reference 2) and February 23, 2010 (Reference 3), Arizona Public Service Company (APS, the licensee), submitted a license amendment request (LAR) to change the Technical Specifications (TSs) for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3. The LAR includes proposed changes that would revise the TSs to relocate the reactor coolant system (RCS) pressure and temperature (P/T) limits and the low temperature overpressure protection (LTOP) enable temperatures to a licensee-controlled document outside of the TSs. The P/T limits and LTOP enable temperatures will be specified in a Pressure and Temperature Limits Report (PTLR) that will be located in the PVNGS Technical Requirements Manual and administratively controlled by a new TS 5.6.9. The licensee's proposed TS revisions were submitted in accordance with U.S. Nuclear Regulatory Commission (NRC) Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 4).

The proposed TSs and the PTLR were documented in Attachments 3 and 4 of Enclosure 1 to Reference 1, respectively. The affected TSs are: TS 1.1, Definitions; TS 3.4.3, RCS Pressure and Temperature (P/T) Limits; TS 3.4.6, RCS Loops – MODE 4; TS 3.4.7, RCS Loops – MODE 5, Loops Filled, TS 3.4.11, Pressurizer Safety Valves – MODE 4, TS 3.4.13, Low Temperature Overpressure Protection (LTOP) System, and TS 5.6.9, Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR).

APS calculated the P/T limits and LTOP system enable temperatures in the proposed PTLR using the methods of topical report Combustion Engineering (CE) NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (Reference 5). This methodology uses a finite element method (FEM) model to calculate the applied stress intensity factors due to pressure, which is not consistent with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, methodology. Therefore, APS also submitted a request for exemption from Title 10 of *the Code of Federal Regulations* (10 CFR) Part 50, Appendix G, "Fracture Toughness Requirements," which specifically cites Section XI, Appendix G, of the ASME Code. The NRC staff's evaluation of that exemption request is the subject of separate correspondence.

The licensee's supplemental letters dated December 22, 2009, and February 23, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 19, 2009 (74 FR 23442).

2.0 REGULATORY EVALUATION

The regulations in 10 CFR Part 50, along with regulatory guidance, are intended, in part, to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The allowable P/T limits to protect the RCPB against brittle failure are defined by P/T curves calculated for normal operations and testing conditions. The LTOP system limits ensure that the pressure remains below the applicable P/T limits. The LTOP system for PVNGS is provided by the relief valves in the suction lines of the shutdown cooling (SDC) system.

The NRC staff has considered the following regulations, regulatory guidance, and generic communications in performing its safety evaluation (SE) of this LAR:

- NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits";

- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements";

- 10 CFR 50.36, "Technical specifications";

- NUREG-1432, "Standard Technical Specifications - Combustion Engineering Plants";

- Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence";

- 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants"; specifically:

  - GDC 14, "Reactor coolant pressure boundary"
  - GDC 15, "Reactor coolant system design"
  - GDC 30, "Quality of reactor coolant pressure boundary"
  - GDC 31, "Fracture prevention of reactor coolant pressure boundary"

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Sections 5.2.2 and 5.3.2; and

- RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

NRC GL 96-03 informed licensees that they may request a license amendment to relocate the actual P/T limit curves and/or LTOP system limit values from the TS limiting conditions for operation (LCOs) into a PTLR or other licensee-controlled document. The GL states that licensees seeking to relocate P/T limits and LTOP system limits for their reactors need to generate their P/T limits and LTOP system limits in accordance with an NRC-approved methodology, and that the methodology used to generate the P/T limits and LTOP system limits needs to comply with the requirements of 10 CFR Part 50, Appendices G and H. Furthermore, the methodology used to generate the P/T limits and LTOP system limits needs to be incorporated by reference in the administrative controls section of the TS. The GL also specifies that the TS administrative controls section for the PTLR must reference the NRC staff's SE approving the use of the PTLR methodology for that facility. The licensee must develop a PTLR or similar document to contain the relocated TS figures, values, and any supporting explanation, and the PTLR must be defined in Section 1.0 of the TS. Attachment 1 to GL 96-03 provides criteria for an approvable methodology; these criteria are discussed in this safety evaluation.

Appendix G to 10 CFR Part 50 requires that facility P/T limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics (LEFM) methodology of Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the ASME Code. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. SRP Section 5.3.2 provides an acceptable method for determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G methodology.

The most recent version of Appendix G to Section XI of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 2004 Edition of the ASME Code. This edition of Appendix G to Section XI of the ASME Code incorporates the provisions of ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," and ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves." Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20 percent of the pre-service hydrostatic test pressure.

The Commission's regulatory requirements related to the content of the TSs are set forth in 10 CFR 50.36. Specifically, 10 CFR 50.36(d)(2)(ii)(B) requires that LCOs be established for a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. In accordance with the 10 CFR 50.36(d)(2)(ii)(B) requirements, LCOs for the P/T limits and LTOP system limits were initially incorporated into the PVNGS TSs. The NRC provided guidance on content and format of TSs for CE plants in

NUREG-1432, "Standard Technical Specifications - Combustion Engineering Plants" (Reference 6), and the staff considered this guidance in its review of the proposed TS changes.

The attributes of an acceptable RPV fluence methodology are described in RG 1.190 (Reference 7). RG 1.190 provides guidance for meeting the requirements of GDC 14, "Reactor coolant pressure boundary," GDC 30, "Quality of reactor coolant pressure boundary," and GDC 31, "Fracture prevention of reactor coolant pressure boundary." GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30 requires that components which are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31 requires that the RCPB be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized.

The review guidance for the LTOP transient analysis is described in SRP Section 5.2.2. SRP Section 5.2.2 provides guidance for meeting the requirements of GDC 15, "Reactor coolant system design," which requires that the RCS be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

3.0 TECHNICAL EVALUATION

The NRC staff's technical evaluation of the licensee's application to relocate the P/T and LTOP limits to a PTLR addresses the proposed methodology, neutron fluence analysis, transient analyses, TS changes, implementation of the PTLR, and the determination of P/T limits.

3.1 Evaluation of the Proposed Methodology for the PTLR

Reference 1 indicated that future changes to the PTLR for PVNGS would be carried out in accordance with an approved version of CE NPSD-683-A (Reference 5). The licensee's proposed PTLR indicated that P/T limits and LTOP system limits were established in accordance with the topical report (TR) documented in Reference 5, which represented the latest NRC-approved version of CE NPSD-683-A and included the NRC SE approving the TR (Reference 8).

The methodology in Reference 5 meets the minimum technical requirements for approved methodologies specified in Attachment 1 to GL 96-03. Since the licensee cited Reference 5 as the proposed methodology for the PTLR for PVNGS, the criterion in GL 96-03 specifying that an approved methodology be used for the development of the PTLR is satisfied. Therefore, the NRC staff concluded that the proposed PTLR for PVNGS is acceptable with respect to this criterion.

3.2 Evaluation of the PTLR Contents

Attachment 1 of GL 96-03 identifies the criteria that the NRC staff will consider in its review of proposed license amendments requesting relocation of P/T limits and LTOP system limits from TS into a PTLR or similar licensee-controlled document. The following discussion provides the

NRC staff's assessment of the licensee's application with respect to addressing the applicable regulatory requirements and criteria in GL 96-03.

3.2.1 Calculation of Neutron Fluence

NRC GL 96-03 specifies that licensees use acceptable methods for reactor vessel neutron fluence calculations (PTLR Criterion 1). APS provided these fluence methodology details and provisions in its February 19, 2009, application as Attachment 5 of Enclosure 1 (Reference 9). The NRC staff evaluated that information to establish that it adheres to the guidance contained in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 7), and is thereby acceptable for GL 96-03 implementation.

The guidance provided in RG 1.190 indicates that the following comprises an acceptable fluence calculation:

- A fluence calculation performed using an acceptable methodology
- Analytic uncertainty analysis identifying possible sources of uncertainty
- Benchmark comparison to approved results of a test facility
- Plant-specific qualification by comparison to measured fluence values

The fast neutron exposure parameters were determined for APS by Westinghouse, using the methodologies discussed in WCAP-16835-NP, "Palo Verde Nuclear Generating Station Units 1, 2, and 3: Basis for RCS Pressure and Temperature Limits Report," and WCAP-16083-NP-A, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry, May 2006" (References 9 and 10). The FERRET methodology has been previously approved by the NRC staff as discussed in Reference 11; the technique discussed in WCAP-16835-NP is addressed in this SE.

For the neutron transport calculations, APS is using the two-dimensional discrete ordinates code, DORT (Reference 12), with the BUGLE-96 cross section library (Reference 13), which was derived from the Evaluated Nuclear Data File (ENDF/B-VI) (Reference 14). Approximations include a P5 Legendre expansion for anisotropic scattering and a S16 order of angular quadrature. These approximations are of a higher order than the P3 expansion and S8 quadrature suggested in RG 1.190. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis for each unit. Three-dimensional flux solutions are constructed using a synthesis of azimuthal, axial, and radial flux. Source distributions include cycle-dependent fuel assembly enrichments, burnups, and axial and radial power distributions, which are used to develop spatial and energy dependent core source distributions that are averaged over each fuel cycle. This method accounts for source energy spectral effects by using an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of each fuel assembly. The neutron transport calculations, as described above, are performed in a manner consistent with the guidance set forth in RG 1.190.

APS determined that the net calculational uncertainty for the fluence calculations was 13 percent, consistent with the RG 1.190 guidance, which specifies that the uncertainty should be less than or equal to 20 percent. APS performed an analytic uncertainty analysis. The calculations were compared with the benchmark measurements from the Poolside Critical Assembly simulator at the Oak Ridge National Laboratory, and with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment. These constitute acceptable test facilities. APS determined the net calculational uncertainty by combining individual components in quadrature.

APS provided, and the NRC staff reviewed, a direct comparison against the measured sensor reaction rates from the PVNGS Unit 2 surveillance program and stated that similar evaluations were performed for Units 1 and 3. For all reactions, the measured-to-calculated (M/C) ratios were very close to unity; the average ratio was 0.97 with 2.8 percent standard deviation. The distribution of M/C ratios ranged from 0.94 to 0.97. Therefore, all reaction rates were calculated within 20 percent of measured values, as recommended in RG 1.190.

The licensee stated that the fluence evaluation contained in Reference 9 provides the calculation of PVNGS reactor vessel neutron fluence projections based on actual plant and fuel cycle calculations to the end of the most recent fuel cycle, and projections to the end of license renewal, or 54 effective full power years (EFPY) of exposure. Therefore, fluence projections through 32 EFPY are acceptable.

The NRC staff reviewed the licensee's neutron fluence calculations, and concluded that the fluence methodology and calculations adhere to the guidance in RG 1.190, and that the licensee has satisfied the provisions of PTLR Criterion 1 from Attachment 1 of GL 96-03.

3.2.2 Calculation of LTOP System Limits

GL 96-03, PTLR Criterion 3, states that,

> Low temperature overpressure protection (LTOP) system limits developed using NRC-approved methodologies may be included in the PTLR. Describe how the LTOP system limits are calculated, applying system thermal-hydraulics and fracture mechanics. Reference SRP Section 5.2.2; ASME Code Case N-514; ASME Code Appendix G, Section XI as applied in accordance with 10 CFR 50.55. Provide setpoint curves or setpoint values.

These specific LTOP analysis methodology details and provisions were reflected in 11 action items associated with the fulfillment of PTLR Criterion 3, which are identified as action items (5) through (15) in Section 5.0 of Reference 8, the NRC SE approving topical report CE NPSD-683 for the LTOP analysis. The staff's evaluation for each of the action items (5) through (15) is provided below.

(5) Provide a description of the analytical method used in the energy addition transient analysis.

Section 3.2.1.3 of WCAP-16385 (Reference 9) provided a description of the analytical method used in the energy addition transient. Since the same method was used in the LTOP energy addition analysis approved by the NRC for a CE plant (Reference 15) and PVNGS is similarly a CE plant, the NRC staff determined that the use of the method in the energy addition analysis is applicable to PVNGS and, therefore, is acceptable.

(6) Provide a description of the analytical method used in the mass addition transient analysis, if different from that in Section 3.3.5 of the topical report.

Section 3.2.1.2 of Reference 9 provided a description of the analytical method used in the mass addition transient. The NRC staff found that the analytical methods used were consistent with those of Section 3.3.5 of the NRC-approved TR (Reference 5) and, therefore, they are acceptable.

(7) Provide a description of the method for selection of relief valve setpoints.

Section 3.2.1.1 of Reference 9 provided a description of the method for selection of relief valve setpoints.

The licensee's analysis of the SDC system relief valve followed the ASME model; the relief valve assumed an initial opening of 30 percent at 3 percent accumulation of the lift setpoint at 481.7 pounds per square inch absolute (psia) (467 pounds per square inch gauge (psig)), and full opening at 10 percent accumulation. The NRC staff found that the relief valve model was consistent with that discussed in Section 3.3.3.3 of the NRC-approved TR (Reference 5), and that the setpoint is adequate for the LTOP system, as demonstrated by the acceptable LTOP analyses discussed in Section 3.3 below. Therefore, the NRC staff concludes that the relief valve model and the associated setpoint are acceptable.

(8) Provide a justification for use of subcooled water conditions or a steam volume in the pressurizer.

Section 3.2.1 of Reference 9 describes initial conditions assumed in the analysis of the mass addition and energy addition events. The analysis assumes that the pressurizer is water-solid with no credit for a cover gas or steam space at the initiation of the events. The NRC staff considers these assumptions to be acceptable.

(9) Provide a justification for a less conservative method for determination of decay heat contribution if the method used is less conservative than the "most conservative method" described in the TR.

Sections 3.2.1 and 3.2.1.3 of Reference 9 discussed the method used to determine the decay heat contribution. In response to the NRC staff's request for additional information, the licensee indicated in Reference 2 that the decay heat curve used in the mass and energy addition transient analysis was based on the 1979 American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1 Standard, "Decay Heat Removal in Light Water Reactors," with 2 sigma uncertainties. Decay heat was maximized for use in the transient analysis by determining the minimum (elapsed) time after reactor shutdown to reach a specific transient temperature during cooldown. Elapsed time after shutdown was determined based on the difference between the RCS cold-leg temperature immediately following a reactor trip and the specific initial temperature of the transient, divided by the maximum allowable cooldown rates, in accordance with the proposed PTLR Table TA2-1 in Reference 16. The NRC staff found that the calculation method of determining the decay heat rates used in the transient analysis conformed to the guidance in Section 3.3.2 of CE NPSD-683-A, and therefore, it is acceptable.

(10) Provide justification for operator action time used in transient mitigation or termination.

As discussed in Section 3.2.1 of Reference 9, the transient analyses in the PVNGS PTLR did not assume operator action for mitigation of the energy or mass addition events discussed. Therefore, the licensee's approach for this item is acceptable.

(11) Provide correlations used for developing Power Operated Relief Valve (PORV) discharge characteristics.

APS did not install pressurizer PORVs at PVNGS Units 1, 2, and 3, and, therefore, did not take credit for PORVs in the LTOP transient analysis, as discussed in Section 3.2.1 of Reference 9.

(12) Provide spring relief valve discharge characteristics if different from those described in the TR or if the peak transient pressure is above the set pressure of the valve plus 10 percent.

Section 3.2.1.1 of Reference 9 provided a description of the PVNGS relief valve discharge characteristics used in the LTOP analyses. As discussed in the evaluation for action item (7) above, the NRC staff determined that the relief valve flow model was consistent with the ASME flow model discussed in Section 3.3.3.3 of the NRC-approved TR, CE NPSD-683-A, Revision 6 (Reference 5).

(13) Provide a description of how the reactor coolant temperature instrumentation uncertainty was accounted for.

Section 5.13 of Reference 9 discussed P/T uncertainties for determination of all RCS P/T limits. A conservative instrument temperature uncertainty of 13.2 degrees Fahrenheit (°F) was added to the analytical temperature limits to increase the allowable temperature limits for the P/T limit values provided in the PVNGS PTLR. The NRC staff finds this assumption acceptable.

(14) Provide a justification for the mass and energy addition transient mitigation which credit presence of nitrogen in the pressurizer.

As discussed in Section 3.2.1 of Reference 9, APS did not credit the presence of nitrogen in the pressurizer in the LTOP transient analysis.

(15) Identify and explain any other deviation from the methodology included in Section 3.0 of the TR.

APS stated that there were no other deviations from the PTLR methodology of Reference 5, with respect to this PTLR criterion, as discussed further in Section 3.3 below.

3.3 <u>LTOP Transient Analyses</u>

TS 3.4.13 ensures that the LTOP system can perform its design safety functions by requiring that at least one of the following overpressure protection systems shall be operable: two SDC system suction line relief valves with lift settings of less than or equal to 467 psig; or the RCS depressurized with an RCS vent greater than or equal to 16 square inches. APS discussed the calculations for determination of the SDC relief valve setpoint for LTOP in Reference 9. The setpoint calculations are done to show the adequacy of the minimum opening pressure of the SDC relief valve to prevent the RCS pressure from exceeding the reactor P/T limits calculated in accordance with the requirements of the applicable regulations. In the setpoint calculations for PVNGS, two types of events were analyzed:

(1) the mass addition event caused by an inadvertent safety injection actuation; and

(2) the energy addition event caused by an inadvertent starting of one inactive reactor coolant pump when a positive steam generator (SG) secondary side water to reactor coolant change in temperature (ΔT) exists.

These events were identified by the licensee as the limiting mass and energy addition events for design of the LTOP system for PVNGS. The following assumptions were used in the mass addition and energy addition analyses in maximizing a peak RCS pressure during the transients:

1. the pressurizer was initially water-solid with no steam space and no credit for the presence of a cover gas (e.g., nitrogen);
2. the RCS boundary did not expand;
3. the RCS metal was adiabatic;
4. the RCS letdown was isolated;
5. all injection pumps reached rated speed instantaneously;
6. only one of two operable SDC relief valves was used in the mitigation;
7. no operator action was credited;
8. full heat output from the pressurizer heaters was assumed; and

2. the SDC system relief valve followed the ASME model; it is assumed that the relief valve initially opens to 30 percent at 3 percent accumulation of the lift setpoint at 481.7 psia (467 psig) and full opening occurs at 10 percent accumulation, at which point each valve is assumed to have a relieving capacity of 5,635 gallons per minute (gpm), which is consistent with that specified in the valve manufacturer's documentation;
3. no reactor coolant pump seal leakage or controlled leak-off was assumed; and
4. the RCS was isothermal and neither heated nor cooled by the mass addition.

3.3.1 LTOP Mass Addition Transient

APS discussed the analysis of the limiting mass addition event in Section 3.2.1.2 of Reference 9. The analysis assumed that an inadvertent actuation signal initiated injection simultaneously from all high pressure safety injection (HPSI) pumps (two) and charging pumps (three) at their design flow rates to the RCS, along with the simultaneous addition of energy from decay heat and the pressurizer heaters. The analysis did not consider mass addition from the safety injection tanks (SITs), since TS 3.5.2 requires the SITs to be isolated during Mode 4 operation with the pressurizer pressure below 430 psia, which is greater than the pressurizer pressure of 385 psia that allows entry into the shutdown cooling mode. The HPSI mass addition was based on the maximum volumetric delivery curves developed for emergency core cooling system calculations using the pressure difference between the RCS and the refueling water tank (RWT). The temperature at the minimum RWT water temperature of 60 °F was assumed for the injected water to establish the greatest rate of mass addition. The method used for the mass addition analysis determined inputs for HPSI mass addition, charging pump mass addition, and the equivalent mass addition that resulted from energy addition. The magnitude of the pressurization was determined by superposition of the mass addition curve onto the relief valve discharge curve, both of which described mass flow rate as a function of pressurizer pressure. The equilibrium pressure was taken as the intersection of the two pressure curves at which the mass addition rate matched the relief valve discharge flow rate.

The NRC staff found that the methods and assumptions used for the mass addition analysis were consistent with the NRC-approved methods discussed in CE NPSD-683-A (Reference 5) and the results of the analysis showed that the peak pressure at the pressurizer during the event remained below 499 psia, which was within the acceptable P/T limits in Table TA2-1, Figures TA2-1 and TA2-2 of the proposed PTLR (Reference 16). Therefore, the NRC staff concluded that the licensee's analysis is acceptable.

3.3.2 LTOP Energy Addition Transient

The licensee analyzed the limiting energy addition event in Section 3.2.1.3 of Reference 9. The licensee assumed startup of one reactor coolant pump at rated speed when the SG secondary side water temperature exceeded the reactor coolant temperature by a ΔT of 100 °F. This ΔT is the maximum value allowed by TS 3.4.13, "Low Temperature Overpressure Protection (LTOP) System," during the LTOP mode. In addition to considering the energy addition to the RCS from the SG secondary side, energy addition from decay heat, the reactor coolant pump and all pressurizer heaters were also included. The analysis was performed with the computer code OVERP. Since the OVERP code was used in an LTOP energy addition analysis approved by

NRC for a CE plant (Reference 11) and PVNGS is also a CE plant, the NRC staff determined that the use of the OVERP code in the energy addition analysis for PVNGS was acceptable. The results of the analysis showed that the peak pressure at the pressurizer during an energy addition event remained below 499 psia, which is within the acceptable P/T limits in Table TA2-1 and Figures TA2-1 and TA2-2 of the proposed PTLR (Reference 16).

The NRC staff found that the licensee's LTOP transient analyses used NRC-approved methodologies for CE plants with acceptable assumptions in maximizing the pressure transient. The results of the analysis showed that the peak RCS pressure at the pressurizer during the limiting transient conditions did not exceed the acceptable P/T limits, meeting the requirements of GDC-15 and GDC-31. The NRC staff, therefore, concluded that the analyses are acceptable.

Based on the discussion in Sections 3.3.1 and 3.3.2 above, the NRC staff concluded that the licensee satisfied the provisions of PTLR Criterion 3 from Attachment 1 of GL 96-03 and associated action items (5) through (15) in the NRC SE of Reference 8.

3.4 Evaluation of Proposed TS Changes

In Attachment 3 of Enclosure 1 to Reference 1, the licensee proposed TS changes related to the relocation of some TS requirements and the implementation of the PTLR for PVNGS. The proposed TS changes affect the definition of the PTLR, certain TS LCOs, related applicability criteria, and administrative controls.

3.4.1 TS 1.1, Definitions

The licensee proposed to add the following definition for the PTLR:

> The PTLR is the site specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.9.

The proposed PTLR definition is modified from the definition in TS 1.1 of NUREG-1432 (Reference 6) by replacing "unit specific document" with "site specific document," to reflect that the single PTLR will contain the P/T limits for all three PVNGS units. Since the P/T limits in the single PTLR were determined for and are applicable to all three units, the NRC staff determined that this change from the NUREG-1432 guidance is editorial in nature. Therefore, the NRC staff concluded that the proposed definition for PTLR meets the intent of the guidance for TS 1.1 of NUREG-1432, which is applicable to CE-manufactured plants including PVNGS Units 1, 2, and 3, and is acceptable.

3.4.2 TS 3.4.3, RCS Pressure and Temperature (P/T) Limits

The licensee proposed to change LCO 3.4.3 to: (1) relocate the RCS P/T limits tables and figures to the PTLR, and reference the PTLR, and (2) relocate the maximum temperature change during hydrostatic testing operations to the PTLR. The NRC staff found that the proposed LCO was identical to LCO 3.4.3 of NUREG-1432, which states that,

> RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

The licensee proposed to change surveillance requirement (SR) 3.4.3.1 to reference the PTLR for the P/T limits and heatup and cooldown rates, and delete the references to TS Table 3.4.3-1, and TS Figures 3.4.3-1 and 3.4.3-2. The NRC staff found that the proposed SR was identical to SR 3.4.1 of NUREG-1432, which requires the operator to "verify RCS pressure, RCS temperature, and heatup and cooldown rates within limits specified in the PLTR" every 30 minutes. The proposed TS also deletes the following TS table and figures:

- Table 3.4.3-1, "Maximum Allowable Heatup and Cooldown Rates"
- Figure 3.4.3-1, "Reactor Coolant System Pressure/Temperature Limitations for Less Than 8 Effective Full Power Years (EFPY) of Operation"
- Figure 3.4.3-2, "Reactor Coolant System Pressure/Temperature Limitations for 8 to 32 Effective Full Power Years of Operation"
- Figure 3.4.3.3, "Maximum Allowable Cooldown Rates for less than 8 EFPY"
- Figure 3.4.3.4, "Maximum Allowable Cooldown Rates for 8 to 32 EFPY"

The NRC staff found that the deleted TS table and figures were included in the proposed PTLR (Reference 16): Table TA2-1 for the RCS heatup and cooldown rate limits; and Figures TA2-1 and TA2-2 for RCS P/T limits for heatup and cooldown, respectively. The deletions make the proposed TS consistent with the proposed relocation of the RCS P/T limits to the PTLR and TS 3.4.3 of NUREG-1432. Therefore, the NRC staff concluded that the proposed changes to TS 3.4.3 are acceptable.

3.4.3 TS 3.4.6, RCS Loops – MODE 4

The licensee proposed to revise TS 3.4.6 LCO Note 2 to reference the PTLR for the LTOP enable temperature. The proposed Note 2 states that,

> No RCP shall be started with one or more of the RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless the secondary side water temperature in each Steam Generator (SG) is < 100°F above each of the RCS cold leg temperatures.

The NRC staff found that the proposed Note was identical to Note 2.b to LCO 3.4.6 of NUREG-1432. Note 2.a of NUREG-1432 addresses conditions when the steam exists in the pressurizer. Since the PVNGS LTOP analyses assumed that the pressurizer was initially water-solid with no steam space and no credit for the presence of a cover gas, the Note 2.a conditions are not applicable to PVNGS and thus, were not included in the proposed TS. Therefore, the NRC staff concluded that the proposed changes to TS 3.4.6 are acceptable.

3.4.4 TS 3.4.7, RCS Loops – MODE 5, Loops Filled

The licensee proposed to revise TS 3.4.7 LCO Note 3 to reference the PTLR for the LTOP enable temperature. The proposed revision to Note 3 states that,

> No Reactor Coolant Pump (RCP) shall be started with one or more of the RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless the secondary side water temperature in each SG is < 100°F above each of the RCS cold leg temperatures.

The NRC staff found that proposed note was identical to Note 3.b to LCO 3.4.6 of NUREG-1432. Note 3.a of NUREG-1432 addresses conditions when steam exists in the pressurizer. Since the PVNGS LTOP analyses assumed that the pressurizer was initially water-solid with no steam space and no credit for the presence of a cover gas, the Note 3.a conditions of NUREG-1432 are not applicable to PVNGS, and it is appropriate that those conditions are not included in the proposed TS. Therefore, the NRC staff concluded that the proposed changes to TS 3.4.7 are acceptable.

3.4.5 TS 3.4.11, Pressurizer Safety Valves – MODE 4

The licensee proposed to revise the TS 3.4.11 applicability statement to state,

> MODE 4 with all RCS cold leg temperatures greater than the LTOP enable temperature specified in the PTLR.

The NRC staff found that the proposed applicability statement is identical to the applicability statement of the corresponding TS 3.4.10 for MODE 4 in NUREG-1432.

The licensee also proposed to revise Required Action A.3 of TS 3.4.11, to reference the PTLR for the LTOP enable temperature. The NRC staff found that the words in the proposed Required Action were identical to the corresponding Required Action B.2 of TS 3.4.10 in NUREG-1432, which requires that the plant,

> Be in MODE 4 with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.

The NRC staff found that the proposed PTLR includes the LTOP enable temperature in Table TA2-2 of Reference 16. Therefore, the NRC staff determined that the proposed changes to TS 3.4.11 are acceptable.

3.4.6 TS 3.4.13, Low Temperature Overpressure Protection (LTOP) System

In the proposed TS 3.4.13 and Table of TA2-2 of the PTLR (Reference 16), the pressure setpoint for the LTOP system to open the SDC relief valve is unchanged from the current TS setpoint of less than or equal to 467 psig, and the enable temperature of the LTOP system is revised from less than or equal to 214°F (as specified in the current TS), to less than or equal to 221 °F (as specified in the proposed PTLR).

The licensee proposed to delete Note 1 for operation with the RCS temperature greater than the LTOP enable temperature, and to revise the applicability statement for MODE 4 of TS 3.4.13, to indicate that the LTOP enable temperature is specified in the PTLR. The NRC staff found that: (1) the proposed deletion of Note 1 is consistent with the corresponding TS 3.4.12 in NUREG-1432, which does not contain Note 1; and (2) the proposed applicability statement is identical to the applicability statement of the corresponding TS 3.4.12 for MODE 4 in NUREG-1432, which states that the LCO is applicable in "MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR." Therefore, the NRC staff determined that the proposed changes to TS 3.4.13 are acceptable.

3.4.7 TS 5.6.9, Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The licensee proposed to add new TS 5.6.9, to specify the appropriate administrative controls to allow the relocation of the P/T limits and LTOP enable temperature to the PTLR, as follows:

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following Technical Specifications (TSs):

    1. TS 3.4.3, RCS Pressure and Temperature (P/T) Limits;
    2. TS 3.4.6, RCS Loops – Mode 4;
    3. TS 3.4.7, RCS Loops – Mode 5 Loops Filled;
    4. TS 3.4.11, Pressurizer Safety Valves – Mode 4; and
    5. TS 3.4.13, Low Temperature Overpressure Protection (LTOP) System.

The NRC staff found that proposed TS 5.6.9.a is consistent with the corresponding TS 5.6.6.a of NUREG-1432, which requires a listing of the individual specifications that address RCS P/T limits that are referenced in the PTLR.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

    - CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.

The PTLR for the PVNGS is documented in Reference 16. Section TA7.0 of the PTLR lists the references to the PTLR, including those NRC-approved methodologies used to determine the P/T limits. Reference 2 of TA7.0 of the PTLR is listed as:

- CE Owners Group Topical Report CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P/T Limits and LTOP Requirements from the Technical Specifications," April 2001.

NRC GL 96-03 allows the relocation of the RCS P/T limits to a PTLR, with appropriate reference in the TS to the NRC-approved TR(s) documenting the methodology for the relocated P/T limits. GL 96-03 specified the inclusion of TR number, title, or other NRC approval document, and date, in the TS for those parameters relocated to the PTLR. The implementation of GL 96-03 is considered acceptable if the administrative controls TS for the PTLR provide a direct link to the specific methodology used to calculate the values previously listed in the TS. In a supplemental submittal, dated February 23, 2010 (Reference 3), the licensee identified CE NPSD-683-A, Revision 6, April 2001, in proposed TS 5.6.9, which provides assurance that only the NRC-approved version of the referenced TR will be used for the determination of the P/T limits and LTOP system limits, since the complete citation (with the TR title, current revision number, and date) is provided. Therefore, the NRC staff found that proposed TS 5.6.9.b specifies the title, revision number, and date for TR CE NPSD-683-A, and, therefore, the proposed TS change is acceptable.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto."

The NRC staff found that proposed TS 5.6.9.c is identical to the corresponding TS 5.6.6.c of NUREG-1432.

The proposed changes to PVNGS TS 1.1, 3.4, and 5.6.9 discussed above correctly reflect the relocation of the RCS P/T limits and LTOP enable temperature to the PTLR, and are consistent with the corresponding standard TSs in NUREG-1432 and the guidance in NRC GL 96-03. The proposed changes also comply with 10 CFR 50.36; therefore, the NRC staff concludes that the proposed TS changes are acceptable.

3.5 PTLR Implementation and P/T Limits

3.5.1 PTLR Implementation

In Reference 1, the licensee provided its evaluation of the proposed PTLR contents against the seven technical criteria in Attachment 1 of GL 96-03 by addressing the 26 action items that were listed in the NRC SE of the CE NPSD-683-A, Revision 6 report, dated March 16, 2001 (Reference 8). Based on its evaluation, the licensee concluded that the proposed PTLR definition (as modified) and controlling TSs meet the technical criteria of GL 96-03 and are consistent with NUREG-1432, Revision 3.1, as modified by NRC-approved Technical Specification Task Force (TSTF) traveler TSTF-408, "Relocation of LTOP Enable Temperature and PORV Lift Setting to the PTLR," dated May 2001.

Since the licensee requested an initial implementation of a PTLR for PVNGS, Units 1, 2, and 3, the NRC staff's review focused on both the implementation of the PTLR and the appropriate application of the CE NPSD-683-A, Revision 6 methodology to generate the proposed P/T limits. The staff examined the licensee's evaluation of the seven technical criteria supporting the PTLR methodology and the proposed PTLR; some of these criteria are discussed in greater detail in Sections 3.1 through 3.3 of this SE. The staff determined that all seven technical criteria are satisfied, as follows:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluences (Sections 1.1 and 1.4 of Reference 9). The proposed PTLR provides the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.

(2) The PTLR methodology describes the surveillance program and identifies the surveillance capsule reports (Section 2.0 of Reference 9). The proposed PTLR provides the reference which contains the surveillance capsule withdrawal schedule. The surveillance capsule reports are also referenced in the proposed PTLR (pages TA-8 and TA-12).

(3) The PTLR methodology describes how the LTOP system limits are calculated, applying system/thermal hydraulics and fracture mechanics.

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2 (Sections 4.2.1, 4.2.2, and 4.2.3 of Reference 9). The proposed PTLR identifies the limiting ART values and limiting materials at the 1/4T and 3/4T locations (page TA-9).

(5) The PTLR methodology describes the application of fracture mechanics in the calculation of P/T limits based on the ASME Code, Section XI, Appendix G, and SRP Section 5.3.2. (Section 5.0 of Reference 9). The proposed PTLR provides the P/T limits for heatup, cooldown, and hydrostatic leak testing (pages TA-6 and TA-7). It should be noted that the licensee's submittal also contains an exemption request, to apply the calculation of flaw stress intensity factors due to membrane stress from pressure loading (K_{IM}) methodology of CE NPSD-683-A, Revision 6, as part of the PVNGS, Units 1, 2, and 3 PTLR methodology. This exemption request will be addressed in separate correspondence.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T limits (Section 5.10 of Reference 9). The proposed PTLR identifies minimum temperatures for the P/T limits, such as minimum bolt-up temperature and hydrostatic test temperature (pages TA-6 and TA-7).

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation. The licensee did not use the surveillance data, because it calculated the ARTs using the table of RG 1.99, Revision 2, and these values are more conservative than those based on the surveillance data (Section 7.0 of Reference 9). The proposed PTLR evaluates the surveillance data in determining that they meet the credibility criteria in RG 1.99, Revision 2 (pages TA-10 to TA-15).

Based on the staff's evaluation of the 7 criteria above, the licensee's implementation of the proposed PTLR is acceptable.

3.5.2 P/T limits

The licensee documented the ART values and P/T limits valid for 32 EFPYs of facility operation using the CE NPSD-683-A, Revision 6 methodology in the proposed PTLR for PVNGS, Units 1, 2, and 3. The licensee identified Intermediate Shell Plate M-6701-2 for the Unit 1 RPV as the limiting material for the heatup and cooldown transients. This material is more limiting than the limiting material of the Unit 2 RPV (Intermediate Shell Plate F-765-6) and the limiting material of the Unit 3 RPV (Lower Shell Plate F-6411-2); therefore, the licensee used the Unit 1 P/T limits as the bounding P/T limits for all three units.

Reference 1 also included the WCAP-16835 report, "Palo Verde Nuclear Generating Station Units 1, 2 and 3; Basis for RCS Pressure and Temperature Limits Report" (Reference 9), which contained detailed information regarding the PVNGS P/T limits. The WCAP-16835 report documented the detailed thermal analyses and fracture mechanics evaluations based on the CE NPSD-683-A, Revision 6 methodology for the PVNGS RPVs, supporting the proposed P/T limits valid for 32 EFPYs. This methodology utilized a 2-dimensional (2-D) FEM model with a postulated inner-diameter (ID) surface flaw at one-quarter of the RPV wall thickness (1/4T) and an FEM model with an outer-diameter (OD) surface flaw at 3/4T to calculate the applied stress intensity factors due to pressure of 1,000 pounds per square inch (K_{IM}). The licensee considered this approach to not be consistent with the ASME Code, Section XI, Appendix G methodology. Therefore, in Reference 1, APS submitted a concurrent request for exemption from 10 CFR 50, Appendix G (which cites ASME Code, Section XI, Appendix G). The applied stress intensity factors due to thermal stresses (K_{IT}) were calculated similarly using the 2-D FEM model. In the final step, the WCAP-16835 report used the applied K_{IT} values and the plane-strain fracture toughness (K_{IC}) values at the crack tip to calculate the allowable pressure stress intensity factor (K_{IP}) at the tip of the postulated flaw at the 1/4T (cooldown) and 3/4T (heatup) locations. Pressure was then obtained by comparing the calculated K_{IP} value to the K_{IM} value based on the FEM model. The resulting P/T limits were further modified to consider minimum bolt-up temperature and the flange limits.

To confirm the licensee's selection of Intermediate Shell Plate M-6701-2 of the Unit 1 RPV as the limiting material for all three PVNGS RPVs under the heatup and cooldown transients, the NRC staff performed ART calculations using the materials and ID fluence information in the NRC's Reactor Vessel Integrity Database (RVID) for PVNGS, Units 1, 2, and 3. The staff found that, except for a minor discrepancy in the chemistry factor for one of the three beltline welds for each of the three PVNGS RPVs, all chemistry factor values, initial reference temperatures (RT_{NDT}) values, and the 1/4T and 3/4T fluence values for other beltline materials of the three PVNGS RPVs in the WCAP-16835 report are identical to those in the RVID. The PVNGS RPVs have credible surveillance data. However, the licensee chose not to use the chemistry factor based on surveillance data, because the chemistry factors based on tables in RG 1.99, Revision 2 are more conservative.

The NRC staff's review of the surveillance data from three capsules indicated that the identification of surveillance specimens is not consistent among the reports. In Reference 2, the licensee confirmed that there was a misidentification of the surveillance specimen for the

Intermediate Shell Plate M-6701-2 in the WCAP-15589 report, "Analysis of Capsule 38° from the Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program (March 2003)." However, this misidentification was corrected in the WCAP-15589, Revision 1 report submitted to NRC by letter dated November 13, 2009 (Reference 17). The staff verified that the licensee used the corrected surveillance data information in its analyses and that the licensee's approach in determining the chemistry factors for beltline materials with varying surveillance data is conservative.

The NRC staff's resulting ART calculations confirmed the licensee's selection of Intermediate Shell Plate M-6701-2 of the PVNGS Unit 1 RPV as the limiting material for all three PVNGS RPVs under the heatup and cooldown transients. The staff did not consider the minor discrepancies between the submittal and the RVID for certain beltline welds to be significant, because those ART values are too low to be limiting. The staff's ART values for the limiting beltline material at the 1/4T and 3/4T locations are 116.3 °F and 102.5 °F for 32 EFPYs, respectively, which are almost identical to the licensee's ART values of 116 °F and 103 °F.

For generating P/T limits, the CE NPSD-683-A, Revision 6 methodology utilized a 2-D FEM model with a postulated ID surface flaw at 1/4T and an FEM model with an OD surface flaw at 3/4T to calculate the K_{IM} values. The NRC staff questioned whether the licensee may have used a generic FEM model without adjusting the results to reflect its plant-specific geometries. In Reference 2, the licensee clarified that, "... Results of the generic 2-D model were then scaled to be specific to the PVNGS reactor vessel." Hence, the staff's concern was resolved. Next, the staff used the licensee's K_{IM} values, the K_{IT} curves in Figure 5-6 (heatup) and Figure 5-8 (cooldown) of Reference 9, and the staff's calculated ART values, to verify the temperature and pressure data in Tables 5-1 and 5-2 of Reference 9. The staff's calculations considered the P/T correction factors, which are presented in Section 5.13 of Reference 9. The staff's verification indicated that for both heatup and cooldown, and for both high and low pressure, the differences between the staff's and the licensee's calculated temperatures are less than 5 percent. This is acceptable agreement, considering the staff's use of hand calculations in some parts of the verification.

Unlike the current P/T limits for PVNGS, the proposed P/T limits do not contain the core critical P/T limit curve. In Reference 2, the licensee stated that the core critical P/T limits are not operationally limiting because the TS LCO 3.4.2, "RCS Minimum Temperature for Criticality," provides a substantial margin to the core critical P/T limits. Therefore, the NRC staff concludes this change is acceptable.

The regulations in 10 CFR Part 50, Appendix G contain additional requirements for the minimum metal temperature of the closure head flange and vessel flange regions. These considerations were reflected in the vertical lines of the proposed P/T limits. The NRC staff verified that when the pressure is greater than 20 percent of the hydrostatic test pressure (514 psia, including pressure uncertainty) the temperature for the hydrostatic test P/T limits is greater than the RT_{NDT} of the limiting flange material plus 90 °F (163.2 °F, including temperature uncertainty), and the temperature for P/T limits when the core is not critical is greater than the RT_{NDT} of the limiting flange material plus 120 °F (193.2 °F including temperature uncertainty). The proposed P/T limits also show a straight line of 80 °F on the low pressure end. This limit was made to meet the 10 CFR Part 50, Appendix G, minimum temperature requirement which limits the operating temperature to the highest RT_{NDT} of the closure flange that is highly stressed by the bolt preload. Since this value for the PVNGS RPVs is 60 °F, the licensee's approach is conservative.

Figures TA2-1 and TA2-2 in the proposed PTLR indicate that some portions of the heatup and cooldown P/T limits overlap (i.e., the P/T curve of another material coexists with the limiting flange curve), and the staff was concerned that the figures could be confusing to the operators. In Reference 2, the licensee clarified that the P/T curves of another material are shown for information, and stated that, "For clarity, the limiting RCS temperatures and RCS heatup and cooldown rate limits are also shown in tabular form in Tables TA2-1 and TA2-2.... Discussions with PVNGS reactor operators confirm that the P/T tables and figures in the PTLR provide clear limits for plant heatup and cooldown operations, in conjunction with the training for licensed operators for this amendment." With this clarification, the NRC staff's concern is resolved.

3.6 Summary of Technical Evaluation

The NRC staff has completed its review of APS's LAR to remove the P/T limit curves and LTOP enable temperature from the PVNGS 1, 2, and 3 TSs and incorporate them into the PTLR, which will be controlled through the implementation of administrative controls specified in new TS 5.6.9. On the basis of its review, the staff has determined the following:

(1) APS has provided fluence calculations performed using an acceptable methodology, supported by analytic uncertainty analysis and comparison to approved test facilities, along with a plant-specific comparison of measured fluence values from the PVNGS reactor vessel surveillance program. Based on these considerations, the staff concludes that APS has followed the guidance in RG 1.190, and the neutron exposures reported in APS's submittal are acceptable. On this basis, the licensee's fluence projections support implementation of NRC GL 96-03.

(2) The LTOP transient analyses used methodologies that were consistent with the methodologies previously approved by NRC for CE plants; the assumptions used in the analyses were acceptable in maximizing the peak pressure during the transients; and the results of the analyses showed that the peak RCS pressure during the limiting transient conditions did not exceed the applicable P/T limits, thereby demonstrating compliance with the requirements of GDC-14, GDC-15, GDC-30, and GDC-31.

(3) The proposed PTLR includes the applicable references that describe the methodologies for the vessel fluence and LTOP transient analyses as specified in GL 96-03; the LTOP transient analyses satisfy the provisions of PTLR Criterion 3 from Attachment 1 of GL 96-03 and associated action items (5) through (15) in PTLR Criterion 3 and Reference 8; the proposed TS changes correctly reflect the proposed relocation of the P/T limits and the LTOP enable temperatures to the PTLR; and the proposed TS changes are consistent with the corresponding TSs in NUREG-1432. Therefore, the proposed PTLR and TS changes are acceptable.

(4) The implementation of the PVNGS PTLR meets the requirements of GL 96-03 because it is developed in accordance with the CE NPSD-683-A, Revision 6 report and APS appropriately addressed all action items stated in the March 16, 2001, SE for the TR.

Therefore, the proposed PVNGS PTLR is approved as part of the PVNGS licensing bases.

(5) The proposed PVNGS RPV P/T limits satisfy the requirements of Appendix G to Section XI of the ASME Code and 10 CFR Part 50, Appendix G, subject to NRC approval of the related exemption request for the calculation of certain stress intensity factors. The NRC staff performed independent evaluations and verified that the proposed P/T limits were developed appropriately using the CE NPSD-683-A, Revision 6 methodology. Therefore, the licensee's proposed P/T limits for PVNGS, Units 1, 2, and 3, are acceptable, and are valid for 32 EFPYs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on May 19, 2009 (74 FR 23442). 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 be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission 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.

7.0 REFERENCES

1. Mims, D.C., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529 and 50-530, Request for Technical Specification Amendment and Exemption from 10 CFR 50, Appendix G, to Relocate the Reactor Coolant System Pressure and Temperature Limits and the Low Temperature Overpressure Protection Enable Temperatures," dated February 19, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090641014).

2. Hesser, J.H., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529 and 50-530, Response to Request for Additional Information for Technical Specification Amendment and Exemption from 10 CFR 50, Appendix G, to Relocate the Reactor Coolant System Pressure and Temperature Limits and the Low Temperature Overpressure Protection Enable Temperatures," dated December 22, 2009 (ADAMS Accession No. ML100040069).

3. Mims, D.C., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529 and 50-530, "Response to Request for Updated Technical Specification Page," dated February 23, 2010 (ADAMS Accession No. ML100560023; Note: this copy of the letter was incorrectly dated February 24, 2010).

4. U.S. Nuclear Regulatory Commission, Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996 (ADAMS Accession No. ML031110004).

5. Combustion Engineering Owners Group and Westinghouse Electric Company LLC, CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," dated April 2001 (ADAMS Accession No. ML011350387).

6. U.S. Nuclear Regulatory Commission, NUREG-1432, Revision 3, "Standard Technical Specifications - Combustion Engineering Plants," dated June 2004 (ADAMS Accession No. ML041830597).

7. U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, dated March 2001 (ADAMS Accession No. ML010890301).

8. Richards, S.A., U.S. Nuclear Regulatory Commission, letter to R. Bernier, Arizona Public Service Company, "Safety Evaluation of Topical Report CE NPSD-683, Revision 6, 'Development of a RCS Pressure and Temperature Limits Report [PTLR] for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications' (TAC No. MA9561)," dated March 16, 2001 (ADAMS Accession No. ML010780017).

9. Westinghouse Electric Company, LLC, "Palo Verde Nuclear Generating Station Units 1, 2, and 3: Basis for RCS Pressure and Temperature Limits Report," WCAP-16835-NP, Revision 0 (Attachment 5 to Enclosure 1 of Reference 1), dated June 2008 (ADAMS Accession No. ML090641016).

10. Westinghouse Electric Company, LLC, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," WCAP-16083-NP-A, dated May 2006 (ADAMS Accession No. ML061600256).

11. Correia, R.P., U.S. Nuclear Regulatory Commission, letter to G. Bischoff, Westinghouse Owners Group, "Final Safety Evaluation Report for Westinghouse Owners Group Topical

Report WCAP-16083-NP, Revision 0, 'Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,'" dated January 10, 2006 (ADAMS Accession No. ML053550466).

12. Radiation Safety Information Computational Center, "Two-Dimensional Discrete Ordinates Transport Code System (DORT)," August 1993.

13. Radiation Safety Information Computational Center, "Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications, (BUGLE-96)" March 1996.

14. McLane, V., et al., "ENDF/B-VI: Evaluated Nuclear Data Library for Nuclear Science and Technology," December 1996.

15. Kalyanam, K., U.S. Nuclear Regulatory Commission, letter to R. Rosenblum, Southern California Edison Company, "San Onofre Nuclear Generating Station, Units 2 and 3 – Issuance of Amendment Re: Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) (TAC Nos. MC5773 and MC5774)," dated July 13, 2006 (ADAMS Accession No. ML062170006).

16. Technical Requirements Manual (Attachment 4 to Enclosure 1 of Reference 1), Appendix TA, "Reactor Coolant System Pressure and Temperature Limits Report (PTLR), Palo Verde Nuclear Generating Station 1, 2, and 3."

17. Weber, T.N., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station (PVNGS) Unit 1, Docket No. STN 50-528, "Revision to Reactor Vessel Material Surveillance Capsule Report," dated November 13, 2009 (ADAMS Accession No. ML093290258).

Principal Contributors: S. Sun
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B. Parks

Date: February 25, 2010

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

James R. Hall, 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:
1. Amendment No. 178 to NPF-41
2. Amendment No. 178 to NPF-51
3. Amendment No. 178 to NPF-74
4. Safety Evaluation

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