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

October 20, 2009

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: REVISION TO TECHNICAL SPECIFICATION 5.5.16, CONTAINMENT LEAKAGE RATE TESTING PROGRAM (TAC NOS. MD9807, MD9808, AND MD9809)

Dear Mr. Edington:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 176 to Facility Operating License No. NPF-41, Amendment No. 176 to Facility Operating License No. NPF-51, and Amendment No. 176 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 October 1, 2008, as supplemented by letters dated July 31 and September 17, 2009.

The amendments modify TS 5.5.16, "Containment Leakage Rate Testing Program," by adding exceptions to the provisions of NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," that would allow the next containment integrated leak rate tests to be performed at 15-year intervals instead of the current 10-year intervals for PVNGS, Units 1, 2, and 3.

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. 176 to NPF-41
2. Amendment No. 176 to NPF-51
3. Amendment No. 176 to NPF-74
4. Safety Evaluation

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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. 176
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 October 1, 2008, as supplemented by letters dated July 31 and September 17, 2009, 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. 176, 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 90 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: October 20, 2009

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. 176
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 October 1, 2008, as supplemented by letters dated July 31 and September 17, 2009, 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. 176, 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 90 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: October 20, 2009

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. 176
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 October 1, 2008, as supplemented by letters dated July 31 and September 17, 2009, 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) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, 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 90 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: October 20, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 176, 176, AND 176

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
5.5-16	5.5.-16

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, 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. 176, 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

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*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. 176, 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.

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

3. The first Type A test performed after the Unit 1 November 1999 Type A test shall be prior to November 4, 2014.

4. The first Type A test performed after the Unit 2 November 2000 Type A test shall be prior to November 2, 2015.

5. The first Type A test performed after the Unit 3 April 2000 Type A test shall be prior to April 27, 2015.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig for Unit 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 58.0 psig for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13. The containment design pressure is 60 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 14.5 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. NPF-51, AND

AMENDMENT NO. 176 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 application dated October 1, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082820029), as supplemented by letters dated July 31 and September 17, 2009 (ADAMS Accession Nos. ML092250649 and ML092670204, respectively), Arizona Public Service Company (the licensee) requested changes to the Technical Specifications (TSs) for Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3. The proposed changes would revise TS 5.5.16, "Containment Leakage Rate Testing Program," by adding exceptions to the provisions of U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. ML003740058), that would allow the next containment (Type A) integrated leak rate test (ILRT) for each unit to be performed at a 15-year interval instead of the current 10-year interval. These changes would allow a one-time, 5-year extension of the ILRT for PVNGS, Units 1, 2, and 3, respectively.

Specifically, the proposed amendment would revise TS 5.5.16.a. to add the following exceptions:

3. The first Type A test performed after the Unit 1 November 1999 Type A test shall be prior to November 4, 2014.

4. The first Type A test performed after the Unit 2 November 2000 Type A test shall be prior to November 2, 2015.

5. The first Type A test performed after the Unit 3 April 2000 Type A test shall be prior to April 27, 2015.

Enclosure 4

The supplemental letters dated July 31 and September 17, 2009, 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 November 18, 2008 (73 FR 68452).

2.0 REGULATORY EVALUATION

Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), requires the licensee to perform an ILRT (a Type A test) and local leakage rate tests (LLRTs), also termed as either Type B or Type C tests. The Type A test measures the overall leakage rate of the primary reactor containment. Type B tests are primarily intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for primary reactor containment penetrations. Type C tests are intended to measure containment isolation valve leakage rates.

The regulations in 10 CFR 50, Appendix J, Option B, "Performance-Based Requirements," require that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Appendix J, Option B, Section V.B.3 states:

> The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or a combined license under part 52 of this chapter to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

2.1 Current PVNGS ILRT Requirements

PVNGS TS Surveillance Requirement (SR) 3.6.1.1 reads, "Perform required visual examinations and leakage rate testing...in accordance with the Containment Leakage Rate Testing Program." TS 5.5.16, "Containment Leakage Rate Testing Program," requires that a testing program be established as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. In its October 1, 2008, letter, the licensee stated that this program shall be in accordance with the guidelines contained in RG 1.163, as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsection IWL, except where relief has been authorized by the Nuclear Regulatory Commission (NRC). The containment concrete visual examination may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWE, except where relief has been authorized by the NRC.

RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995. NEI 94-01, Revision 0, paragraph 9.2.3, "Extended Test Intervals," specifies that, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history."

Regulatory Position C.1 of RG 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01, which references Section 9.0. Paragraph 9.2.3 defines acceptable performance history as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 La (where La is the maximum allowable leakage rate at the calculated peak containment internal pressure related to the design-basis loss-of-coolant accident, as specified in the TSs). The PVNGS reactor containment vessels have met this criterion and, therefore, currently qualify for the 10-year frequency.

TS 5.5.16.b states,

> The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 52.0 psig [pounds per square inch gauge] for Unit 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 58.0 psig for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13. The containment design pressure is 60 psig.

All three PVNGS units have completed their associated operating cycle 13, which included steam generator replacement and power uprates. Therefore, the peak calculated containment pressure, Pa, is now 58.0 psig for all units.

TS 5.5.16.c states,

> The maximum allowable containment leakage rate, La, at Pa, shall be 0.1% of containment air weight per day.

The maximum allowable containment leakage rate, La, specified in TS 5.5.16, ensures that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. As an added conservatism to account for possible degradation of the containment leakage barriers between leakage tests, TS 5.5.16.d limits the leakage rate acceptance criteria as follows:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance [criteria] are < 0.60 La for the Type B and C tests and $\leq$ 0.75 La for Type A tests.

3.0 TECHNICAL EVALUATION

The licensee submitted information to support the proposed changes to extend the current ILRT intervals for PVNGS, Units 1, 2, and 3. The supporting information includes historical plant-specific leak rate test performance, the Containment In-service Inspection (CISI) results, and a risk-informed analysis. The NRC staff reviewed the licensee's technical analysis in its submittal dated October 1, 2008, and in its supplemental letters dated July 31 and September 17, 2009, which responded to the staff's requests for additional information (RAI).

This NRC safety evaluation addresses the licensee's assessment of the current condition of the structural and leak-tight integrity of the containment; the adequacy of the licensee's ILRT, LLRT, and CISI programs to detect and manage degradation of the containment; and the licensee's risk impact assessment for the proposed ILRT interval extension.

3.1 Containment Building Description

PVNGS's Updated Final Safety Analysis Report (UFSAR), Section 1.2.12.1, Containment Building, describes the containment building as follows:

> The containment building is a prestressed concrete cylinder with a hemispherical dome. The basemat is a flat, circular slab of reinforced concrete. The interior of the structure is lined with a continuous, welded steel plate ¼ inch thick. Approximate dimensions of the structure are:

Structure Characteristic	Dimensions (ft.)
Inside diameter	146
Inside height	206.5
Vertical wall thickness	4
Dome thickness at apex	3.5
Basemat diameter	161
Basemat thickness	10.5
Net Free Volume	2.62×10^6 ft^3

> The Containment building is designed for a maximum internal pressure of 60 psig and a maximum, accident condition inner surface temperature of 300 degrees F. Housed within the containment building and supported by the basemat are the reinforced concrete and structural steel internal structures that support the reactor and reactor coolant system.

> Under the most severe of postulated loading conditions -- including the combined effects of permanent loads, design basis LOCA [loss of coolant accident] loads, and either the safe shutdown earthquake or tornado loads -- the containment building is designed to maintain its structural and leak tight integrity. This design permits a predictable response of the containment structure to allow operation of engineered safety features equipment for mitigation of accident consequences. Together with isolation valves, penetration assemblies, and its continuous,

welded steel liner, the structure contains the released fission products and maintains a leak rate below the design leak rate levels. The containment is designed to provide long-term control of fission products following a postulated accident.

Containment penetrations are provided in the lower portion of the structure and consist of a personnel airlock, an equipment hatch, an emergency airlock, a fuel transfer tube, and piping, electrical, instrumentation, and ventilation penetrations.

As discussed in UFSAR Section 3.8.1.1.3.1, Liner Plate and Anchors, a welded steel liner plate covers the entire inside surface of the containment (excluding penetrations) to satisfy the leak-tight criteria. The liner is typically ¼-inch thick and is thickened locally around penetration sleeves, large brackets, and attachments to the basemat and shell wall.

As discussed in UFSAR Section 3.8.1.1.1, Containment Basic Configuration, the ¼-inch thick containment liner which runs on top of the basemat is covered by a 33-inch thick concrete filler slab that supports the containment internals and forms the floor of the containment. The filler slab is an internal structure and is not within the scope of the ILRT program.

The containment pressure boundary consists of the steel liner, containment access penetrations, and penetrations for process piping and electrical services. The integrity of the penetrations and containment isolation valves is verified through Type B and Type C tests, and the overall integrity of the containment structure is verified through Type A tests, as required by 10 CFR Part 50, Appendix J. These tests are performed to verify the leak-tight integrity of the containment structure at the design-basis accident pressure. The leakage rate testing requirements of 10 CFR 50, Appendix J, Option B (Type A, Type B, and Type C tests) and the CISI requirements mandated by 10 CFR 50.55a, "Codes and standards," together, ensure the continued leak-tight and structural integrity of the containment during its service life.

3.2 Historical Testing Results

The licensee provided summaries of the historical ILRTs and the combined (total) Type B and Type C testing as-found minimum pathway results calculated at each refueling outage back to the year 2002 for all three units at PVNGS. All ILRT results were less than 67 percent of La (performance criterion is 75 percent of La), with no apparent adverse trends that would suggest containment leakage potential would exceed La during the requested 5-year interval extensions. The Type B and C test minimum pathway totals were all less than 6 percent of their performance criterion (0.6 La) and were also without any apparent adverse trend to suggest containment leakage potential would exceed La during the requested 5-year interval extensions. The Type B and Type C testing schedules are expected to be minimally impacted by the requested ILRT extensions, and these tests will continue to be performed and the results totaled each refueling outage. Penetration leakage is expected to be the major contributor of any potential containment leakage and the Type B and Type C tests will continue to provide monitoring of potential penetration leakage at the existing allowed intervals for these tests.

3.2.1 PVNGS, Units 1, 2, and 3 Type A Tests

The licensee provided the following results of previous Type A test results and the leakage rate acceptance criteria in its supplemental letters dated July 31 and September 17, 2009.

Unit 1 Type A Test Results:

Test Date	December 1982	May 1986	February 1990	November 1999	Acceptance Criteria
Total As-Found Leakage	0.14 La	0.66 La	0.67 La	0.58 La	≤ 1.0 La
Total As-Left Leakage	0.14 La	0.66 La	0.66 La	0.55 La	≤ 0.75 La

Unit 2 Type A Test Results:

Test Date	February 1985	June 1988	December 1991	November 2000	Acceptance Criteria
Total As-Found Leakage	0.09 La	0.6 La	0.83 La	0.42 La	≤ 1.0 La
Total As-Left Leakage	0.09 La	0.6 La	0.31 La	0.42 La	≤ 0.75 La

Unit 3 Type A Test Results:

Test Date	September 1986	April 1991	April 2000	Acceptance Criteria
Total As-Found Leakage	0.52 La	0.64 La	0.51 La	≤ 1.0 La
Total As-Left Leakage	0.52 La	0.62 La	0.51 La	≤ 0.75 La

"La," is 0.1% of containment air weight per day.

The results of the Type A tests show that the containment leakage is within the established acceptance limits with adequate margin, which provides reasonable assurance of leak-tightness of the PVNGS Units 1, 2, and 3 containment structures.

Regulatory Position C.3 of RG 1.163 recommends that a visual examination of accessible interior and exterior surfaces of the containment structure should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test, based on a 10-year ILRT interval. The NRC staff requested the licensee to describe the plan to supplement the 10-year interval-based visual inspection requirement to accommodate the requested 15-year ILRT interval. In response to the staff's RAI, the licensee stated that the visual inspection requirements are addressed by inspections performed in accordance with the ASME Code, Section XI, Subsections IWE and IWL programs. In addition to the visual inspection of the concrete surfaces of the containment structure performed as part of IWL examinations, the licensee committed to perform a supplemental general visual examination of the accessible exterior surfaces of the containment structure. Considering this supplemental general visual examination, as summarized in the table below with approximate dates for future examinations, there will be a total of three visual examinations of the accessible exterior surfaces of the containment structure prior to the pre-ILRT general visual examination.

	Visual Examination of Containment Exterior Surfaces			
	1st IWL Inspection (Baseline)	**2nd IWL Inspection**	**Supplemental General Visual Exam**	**Pre-ILRT General Visual Exam**
Unit 1	September 2001	September 2006	2011	October 2014
Unit 2	September 2001	September 2011	2013	October 2015
Unit 3	September 2001	September 2011	2013	April 2015

Furthermore, the licensee stated that the first 10-year interval of the IWE examination has been completed. In its July 31, 2009, letter, the licensee provided the schedule for the second 10-year interval for the IWE examination of the accessible interior surfaces of the PVNGS, Units 1, 2, and 3 containment structures, as shown in the table below:

	Visual Examination of Containment Interior Surfaces			
	2nd IWE inspection Interval - 1st period	**2nd IWE Inspection Interval - 2nd period**	**2nd IWE Inspection Interval - 3rd period**	**Pre-ILRT General Visual Exam**
Unit 1	July 2008 to November 2011	November 2011 to March 2015	March 2015 to July 2018	October 2014
Unit 2	March 2007 to July 2010	July 2010 to November 2013	November 2013 to March 2017	October 2015
Unit 3	January 2008 to May 2011	May 2011 to September 2014	September 2014 to January 2018	April 2015

Considering the extended 15-year ILRT interval dates for PVNGS, Units 1, 2, and 3, at least two visual examinations of the accessible interior surfaces of the containment structure will be performed, as part of the second IWE inspection interval, prior to the pre-ILRT general visual examination. Also, at least one visual examination of the accessible interior surfaces of the containment structure has been completed during the first IWE inspection interval. Therefore, there will be at least a total of three visual examinations of the accessible interior surfaces of the containment structure prior to the pre-ILRT general visual examination.

The NRC staff concludes the licensee's plan to perform at least three visual examinations of the accessible interior and exterior surfaces of the containment structure, prior to the pre-ILRT visual examination for the extended 15-year ILRT interval, is consistent with the intent of Regulatory Position C.3 of RG 1.163 and is, therefore, acceptable.

3.2.2 PVNGS, Units 1, 2, and 3 Type B and C Tests

As stated in the licensee's amendment request, the TS requirements for Type B and Type C testing of containment penetrations and isolation valves will not be affected by the extension of the Type A test interval. In its July 31, 2009, response to the NRC staff's RAI, the licensee provided the as-found combined leakage rates for Type B and Type C tests performed during the refueling outage R10 (October 2002 for Unit 1, April 2002 for Unit 2, and May 2003 for

Unit 3), through refueling outage R14 (November 2008 for Unit 1, May 2008 for Unit 2, and May 2009 for Unit 3). These test results were all in compliance with the TS acceptance limits.

The licensee stated, in the October 1, 2008, amendment request, that expansion bellows are not utilized in the design of the mechanical penetrations at PVNGS. However, there are bellows used on the fuel transfer tube penetration to accommodate the relative movement between the refueling canal liner and the containment building penetration. These bellows do not form part of the containment building vessel nor the pressure boundary, and they are unaffected by this proposed license amendment. Based on its responses to the NRC staff's RAI, the licensee clarified that the fuel transfer tube penetration is designated as containment penetration 53 and the blind flange on this penetration receives a Type B test every refueling outage. The licensee further stated that to date there is no PVNGS operating experience that indicates leakage through the fuel transfer tube penetration pipe wall, welds, or bellows.

In its July 31, 2009, response to the staff's RAI, the licensee provided a summary table of the future Type B and Type C tests to be performed prior to and during the requested 5-year extension of the ILRT interval.

Based on the above, the NRC staff finds that the integrity of the containment pressure boundary is effectively monitored through the Type B and Type C testing of the containment penetrations and isolation valves, respectively, as required by 10 CFR 50, Appendix J, and the PVNGS TS.

3.2.3 Containment In-Service Inspection

In the October 1, 2008, amendment request, the licensee stated that: (1) the IWE and IWL related CISI program is unaffected by the proposed TS amendment; (2) the first IWL baseline inspection of PVNGS, Units 1, 2, and 3 containment structures was completed in September 2001, and these examinations uncovered no evidence of containment degradation; (3) the 2007 IWL inspection of the Unit 1 containment found no conditions that would impact its structural integrity; (4) the first 10-year interval IWE examination of the PVNGS, Units 1, 2, and 3 containments has been completed and all results were within the established acceptance criteria; and (5) there are currently no areas/locations subject to augmented inspection at PVNGS, Units 1, 2, and 3.

Under its CISI program, as required by 10 CFR 50.55a(b)(2)(viii)(E) and (b)(2)(ix)(A), the licensee evaluates the acceptability of inaccessible areas of the containment structure and metallic liner, if conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. In its July 31, 2009, response to the NRC staff's RAI, the licensee stated that, to date, there have been no conditions in PVNGS, Units 1, 2, and 3 that have indicated the presence of, or resulted in degradation to, the inaccessible areas.

Regarding the moisture barrier at the interface of the containment liner and the containment concrete floor, the licensee stated that a moisture barrier is not utilized in the design of PVNGS. One hundred percent of the accessible areas of the containment floor, including areas that might allow water to penetrate to the liner plate below, are inspected each inspection period as part of the IWE program. The licensee also stated that, as part of the component monitoring program, the containment liner protective coating is inspected visually every refueling outage to verify coating thickness and condition.

On the basis of its review of the information provided in the licensee's amendment request and its responses to the NRC staff's RAI, the staff finds that: (1) the results of the past ILRTs demonstrate that the leak-tight integrity of the containment structure has been adequately managed; (2) the containment leak-tight integrity is verified through periodic CISI conducted as required by Subsections IWE and IWL of the ASME Code, Section XI; (3) the leak-tight integrity of the containment penetrations, containment isolation valves, airlocks, and seals and gaskets are periodically verified through the Type B and Type C tests, as required by 10 CFR Part 50 Appendix J, and the PVNGS, Units 1, 2, and 3 TSs; (4) the licensee is employing a CISI program that requires evaluation of any potential degradation of accessible and inaccessible areas of the containments; and (5) the containment liner protective coating is inspected visually every refueling outage and any identified damage is repaired, as necessary. Based on the above, the NRC staff concludes that the licensee's proposed one-time extension of the ILRT interval from 10 to 15 years for PVNGS, Units 1, 2, and 3, is acceptable.

3.3 Risk Analysis

The licensee performed a risk impact assessment of extending the Type A test interval from 10 to 15 years. The risk assessment was provided in the October 1, 2008, application for license amendment. Additional analysis and information were provided by the licensee in its letter dated July 31, 2009, in response to the NRC staff's RAI. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) topical report (TR)-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," August 1994, and NRC RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998 (ADAMS Accession No. ML003740133). The licensee also performed its risk assessment with consideration of the guidance in the NEI/EPRI report, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Surveillance Intervals," dated November 2001, and the methodology in EPRI TR-1009325, Revision 2, Final Report, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, the industry undertook a similar study; the results of that study are documented in EPRI TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for PVNGS early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests, based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of pre-existing containment leakage. The risk contribution of pre-existing leakage for the pressurized-water reactor and boiling-water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person roentgen equivalent man (rem) per year in increased public dose.

The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak was present. Since the Option B rulemaking was completed in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 states that a PRA used in risk-informed regulation should be performed in a manner that is consistent with accepted practices. In NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated January 2007 (ADAMS Accession No. ML070240001), to determine whether the technical adequacy of the PRA used to support a submittal is consistent with accepted practices. Revision 2 of RG 1.200 will be used for all risk-informed applications received after March 2010. In the Final Safety Evaluation for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML081140105), the NRC staff states that Capability Category I of the ASME PRA Standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their contribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

In accordance with this guidance, the licensee's October 1, 2008, license amendment request and July 31, 2009, letter address the technical adequacy of the PRA that forms the basis for the subject risk assessment. An industry peer review team reviewed an older version of the PRA model in November 1999. In 2001, a contractor review of the licensee's responses to the peer review findings determined that there was only one open fact and observation (F&O) from the 1999 peer review. As part of the ILRT extension application and in response to NRC staff's RAI, the licensee reported the results of a self-assessment of its current PRA model (Revision 14) at the time of the application to evaluate conformance with RG 1.200 Capability Category I and II guidance. A summary of the findings from the self-assessment, and an assessment of the impact of these findings on the risk assessment for the ILRT extension, are provided in the licensee's October 1, 2008, license amendment request and July 31, 2009, letter. The one open F&O from the 1999 peer review is the lack of an internal flood (IF) model. The licensee's assessment stated that the screening process used in the Individual Plant Examination of External Events (IPEEE) had screened out all compartments based on risk, that the screening process used in the IPEEE is consistent with more recent IF guidelines, that no new information since the IPEEE contradicts these findings, and that the risk contribution of internal flooding is expected to be minimal. The licensee further stated that the model changes required to address the remaining findings would have a negligible, if any, impact on the results of the risk assessment. The NRC staff reviewed this information and has no objection to the conclusions in the licensee's assessment. Given that the licensee has evaluated its PRA against RG 1.200 and the ASME PRA Standard, evaluated all of the findings developed during the reviews of its PRA for applicability to the ILRT extension, and determined that any unresolved issues would not impact the conclusions of the ILRT risk assessment, the NRC staff concludes that the current PVNGS PRA model is of sufficient technical quality to support the evaluation of changes to ILRT frequencies.

RG 1.174 provides risk-acceptance guidelines for assessing the increases in CDF and LERF for risk-informed license amendment requests. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed amendment based on the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth. The licensee estimated the change in the conditional containment failure probability for the proposed amendment and judged it to be insignificant and reflecting sufficient defense-in-depth.

The licensee comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be less than 0.001 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be about 5.9×10^{-9} per year, based on the plant-specific internal events PRA, and about 1.3×10^{-7} per year, when external events are included. There is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in ASME Code, Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 10^{-8} per year, including external and internal events.

Pursuant to RG 1.174, when the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered if the total LERF is less than 10^{-5} per year. Based on information provided by the licensee, the total LERF for internal and external events, including the requested change, is about 1.6×10^{-6} per year, which meets the total LERF criterion. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also discusses the need to show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of less than one percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff concludes that the increase in projected risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy of RG 1.174, and is, therefore, acceptable.

Based on the above, the NRC staff concludes that the proposed license amendment request for a one-time, 5-year extension of the Type A containment integrated leak rate test interval for the PVNGS, Units 1, 2, and 3, is acceptable. In accordance with revised TS 5.5.16, the next Type A tests for PVNGS, Units 1, 2, and 3, shall be performed no later than November 4, 2014, November 2, 2015, and April 27, 2015, respectively.

4.0 REGULATORY COMMITMENTS

As stated in its letter dated September 17, 2009, the licensee committed to perform a supplemental general visual examination of the accessible exterior surfaces of the PVNGS, Units 1, 2, and 3 containment structures. This supplemental examination is in addition to the regularly scheduled IWL inspection of the containment structure and it will be performed for each unit prior to the pre-ILRT general visual examination during the requested extended ILRT interval. The NRC staff has identified this as a regulatory commitment.

5.0 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.

6.0 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 November 18, 2008 (73 FR 68452). 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.

7.0 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.

Principal Contributors: F. Farzam
J. Bettle
T. Ghosh

Date: October 20, 2009

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. 176 to NPF-41
2. Amendment No. 176 to NPF-51
3. Amendment No. 176 to NPF-74
4. Safety Evaluation

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NAME	RElliott CSchulten for	BMizuno	MMarkley	JRHall	
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