

September 11, 1997

Mr. James M. Levine
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

SUBJECT: ISSUANCE OF AMENDMENTS FOR THE PALO VERDE NUCLEAR GENERATING STATION
UNIT NO. 1 (TAC NO. M97524), UNIT NO. 2 (TAC NO. M97525), AND UNIT
NO. 3 (TAC NO. M97526)

Dear Mr. Levine:

The Commission has issued the enclosed Amendment No. 113 to Facility Operating License No. NPF-41, Amendment No. 106 to Facility Operating License No. NPF-51, and Amendment No. 85 to Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated December 27, 1996, as supplemented by letter dated August 22, 1997.

These amendments change Technical Specification 3/4.6.1.3.b and its associated Bases sections to reflect an increase in the peak containment internal pressure for the design basis loss-of-coolant accident (LOCA) from 49.5 psig to 52 psig.

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,
Original Signed By
Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

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

- Enclosures:
1. Amendment No. 113 to NPF-41
 2. Amendment No. 106 to NPF-51
 3. Amendment No. 85 to NPF-74
 4. Safety Evaluation

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DOCUMENT NAME: PV97524.AMD

OFC	PDIV-2/PM	PDIV-2/LA	SCSB/BC <i>CHB</i>	OGC
NAME	<i>KMT</i> KThomas:ye	<i>ESP</i> EPeyton	CBerlinger	<i>APH</i>
DATE	8/26/97	8/26/97	8/26/97 <i>BW</i>	8/26/97

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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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
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 December 27, 1996, as supplemented by letter dated August 22, 1997, 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. 113, 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 to be implemented within 30 days from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kristine M. Thomas

Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 11, 1997

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. NPF-41

DOCKET NO. STN 50-528

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE

3/4 6-4
B 3/4 6-2
6-25

INSERT

3/4 6-4
B 3/4 6-2
6-25

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a , 52 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days, or
 2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
 3. The provisions of Specification 3.0.4 are not applicable.
- b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. In accordance with the Containment Leakage Rate Testing Program at periodic intervals and following each closing as specified. The provisions of Specification 4.0.2 are not applicable to the overall air lock leakage test.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L, or less than or equal to 0.75 L, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Option B of Appendix J of 10 CFR Part 50 and Regulatory Guide 1.163, dated September 1995.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 4 psig and (2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA conditions.

The maximum pressure increase expected to be obtained from a LOCA event is 49.5 psig. The limit of 2.5 psig for initial positive containment pressure will limit the total pressure to 52 psig which is less than the design pressure (60 psig) and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 52 psig in the event of a LOCA. The containment design pressure is 60 psig. The measurement of containment tendon lift-off force; the tensile tests of the tendon wires or strands; the examination and testing of the sheathing filler grease; and the visual examination of tendon anchorage assembly hardware, surrounding concrete and the exterior surfaces of the containment are sufficient to demonstrate this capability. The tendon wire or strand samples will also be subjected to tests. All of the required testing and visual examinations should be performed in a time frame that permits a comparison of the results for the same operating history.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 1, 1974.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

ADMINISTRATIVE CONTROLS

- 2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
- 3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;
- 4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
- 5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;
- 6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; and
- 7) An estimate of the exposure to plant operating personnel as a result of the change.

6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR Part 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 and ANSI/ANS-56.8-1994.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 52 psig.

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

Leakage rate acceptance criteria are:

- a. 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 criteria are $\leq 0.6 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when pressurized to $\geq 14.5 \pm 0.5$ psig.



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 NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
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 December 27, 1996, as supplemented by letter dated August 22, 1997, 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-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. 106, 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 to be implemented within 30 days from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kristine M Thomas

Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 11, 1997

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-51

DOCKET NO. STN 50-529

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE

3/4 6-4
B 3/4 6-2
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INSERT

3/4 6-4
B 3/4 6-2
6-25

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a , 52 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days, or
 2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
 3. The provisions of Specification 3.0.4 are not applicable.
- b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. In accordance with the Containment Leakage Rate Testing Program at periodic intervals and following each closing as specified. The provisions of Specification 4.0.2 are not applicable to the overall air lock leakage tests.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L, or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Option B of Appendix J of 10 CFR Part 50 and Regulatory Guide 1.163, dated September 1995.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 4 psig and (2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA conditions.

The maximum pressure increase expected to be obtained from a LOCA event is 49.5 psig. The limit of 2.5 psig for initial positive containment pressure will limit the total pressure to 52 psig which is less than the design pressure (60 psig) and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 52 psig in the event of a LOCA. The containment design pressure is 60 psig. The measurement of containment tendon lift-off force; the tensile tests of the tendon wires or strands; the examination and testing of the sheathing filler grease; and the visual examination of tendon anchorage assembly hardware, surrounding concrete and the exterior surfaces of the containment are sufficient to demonstrate this capability. The tendon wire or strand samples will also be subjected to tests. All of the required testing and visual examinations should be performed in a time frame that permits a comparison of the results for the same operating history.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 1, 1974.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

ADMINISTRATIVE CONTROLS

- 1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.
- 2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
- 3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;
- 4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
- 5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;
- 6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; and
- 7) An estimate of the exposure to plant operating personnel as a result of the change.

6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR Part 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 and ANSI/ANS-56.8-1994.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 52 psig.

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

Leakage rate acceptance criteria are:

- a. 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 criteria are $\leq 0.6 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.

ADMINISTRATIVE CONTROLS

6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM (Continued)

- 2) For each door, leakage rate is $\leq 0.01 L_s$ when pressurized to $\geq 14.5 \pm 0.5$ psig.



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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
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 December 27, 1996, as supplemented by letter dated August 22, 1997, 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-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. 85, 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 to be implemented within 30 days from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kristine M Thomas

Kristine M. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 11, 1997

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-74

DOCKET NO. STN 50-530

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE

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INSERT

3/4 6-4
B 3/4 6-2
6-25

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a , 52 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days, or
 2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
 3. The provisions of Specification 3.0.4 are not applicable.
- b. With the containment air lock inoperable, except as the result of an inoperable air lock door maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. In accordance with the Containment Leakage Rate Testing Program at periodic intervals and following each closing as specified. The provisions of Specification 4.0.2 are not applicable to the overall air lock leakage tests.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75 L$, or less than or equal to $0.75 L_t$, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Option B of Appendix J of 10 CFR Part 50 and Regulatory Guide 1.163, dated September 1995.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 4 psig and (2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA conditions.

The maximum pressure increase expected to be obtained from a LOCA event is 49.5 psig. The limit of 2.5 psig for initial positive containment pressure will limit the total pressure to 52 psig which is less than the design pressure (60 psig) and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 52 psig in the event of a LOCA. The containment design pressure is 60 psig. The measurement of containment tendon lift-off force; the tensile tests of the tendon wires or strands; the examination and testing of the sheathing filler grease; and the visual examination of tendon anchorage assembly hardware, surrounding concrete and the exterior surfaces of the containment are sufficient to demonstrate this capability. The tendon wire or strand samples will also be subjected to tests. All of the required testing and visual examinations should be performed in a time frame that permits a comparison of the results for the same operating history.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of UngROUTED Tendons in Prestressed Concrete Containment Structures," Revision 1, 1974.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

ADMINISTRATIVE CONTROLS

- 1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.
- 2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
- 3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;
- 4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
- 5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;
- 6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; and
- 7) An estimate of the exposure to plant operating personnel as a result of the change.

6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR Part 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 and ANSI/ANS-56.8-1994.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 52 psig.

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

Leakage rate acceptance criteria are:

- a. 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 criteria are $\leq 0.6 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.

6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM (Continued)

- 2) For each door, leakage rate is $\leq 0.01 L_0$ when pressurized to $\geq 14.5 \pm 0.5$ psig.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. NPF-41,
AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-51,
AND AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-74
ARIZONA PUBLIC SERVICE COMPANY, ET AL.
PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3
DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By application dated December 27, 1996, as supplemented by letter dated August 22, 1997, the Arizona Public Service Company (APS or the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, respectively) for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3. The Arizona Public Service Company submitted this request on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority. The proposed changes would revise Technical Specification (TS) 3/4.6.1.3.b and the associated Bases sections (3/4.6.1.4, 3/4.6.1.6 and 6.16) to reflect an increase in the peak containment internal pressure for the design basis LOCA from 49.5 psig to 52 psig.

The August 22, 1997, supplemental letter provided additional clarifying information that did not change the staff's original no significant hazards consideration determination that was published in the Federal Register on May 21, 1997 (62 FR 27794).

2.0 EVALUATION

Technical Specification 3.6.1.3.b provides the limits on containment air lock leakage based on the calculated peak containment internal pressure for the design basis LOCA. The limits on leak rates for containment air locks are required to meet the restrictions for containment integrity and containment leak rate. The licensee indicated that a reanalysis using the 2.5 psig initial positive containment pressure limit resulted in a change to the calculated peak containment pressure for the design basis LOCA from 49.5 psig to 52.0 psig. The proposed change is still below the containment design pressure of 60.0 psig. In addition, it maintains at least a 10 percent margin

above the accepted peak calculated containment pressure following a LOCA as per NRC Standard Review Plan (SRP), Section 6.2.1.1.A, "PWR Dry Containment, Including Subatmospheric Containments." With this change, the containment will still meet the affected General Design Criteria 16, 38, and 50 of 10 CFR 50, Appendix A. Further, the change will not invalidate the Acceptance Criteria "a" through "i" specified in Section 6.2.1.1.A of the SRP. The licensee also stated that this change does not impact the radiological consequences of a LOCA as previously analyzed in Chapters 6 and 15 of the Updated Safety Analysis Report (UFSAR). UFSAR Section 15.6.5.6 "Analyses of Effects and Consequences - Large Break LOCA," states that "it is assumed that the containment leaks at the maximum rates allowed by the Technical Specifications, i.e., 0.1 vol.%/d for the first 24 hours and half of that rate thereafter." The dose calculation assumes that under accident conditions, the release of radionuclides to the containment is instantaneously mixed with containment air within the containment free air volume. This results in a constant radioactivity per volume (curies/cc) independent of containment internal pressure. Since radioactivity is assumed to be mixed in the containment free air volume, the volume percent leaked per day is equivalent to the fraction of radioactivity which leaks from the containment per day. Therefore, the increase in the peak calculated containment internal pressure does not impact the radiological consequences associated with the design basis LOCA as analyzed in Chapters 6 and 15 of the UFSAR.

Based on the above, the staff finds the proposed change for the peak containment internal pressure for the design basis LOCA to be acceptable as this value remains below the containment design pressure and still meets the 10 percent margin. Further, it does not impact the radiological consequences of a LOCA previously analyzed.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the 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 (62 FR 27794). 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.

5.0 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.

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Date: September 11, 1997