

1996  
February 23, 1996

Mr. William L. Stewart  
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. M94295), UNIT NO. 2 (TAC NO. M94296),  
AND UNIT NO. 3 (TAC NO. M94297)

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 103 to Facility Operating License No. NPF-41, Amendment No. 92 to Facility Operating License No. NPF-51, and Amendment No. 75 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 (TS) in response to your application dated December 19, 1995, as supplemented by letter dated February 9, 1996.

These amendments would allow the implementation of the recently approved Option B to 10 CFR Part 50, Appendix J. This new rule allows for a performance-based option for determining the test frequency for containment leakage rate testing. The proposed amendment would modify TS 1.7, 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, and 3/4.6.3, and the Bases of TS 3/4.6.1.2, and would add a new TS 6.16.

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

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

- Enclosures: 1. Amendment No. 103 to NPF-41  
2. Amendment No. 92 to NPF-51  
3. Amendment No. 75 to NPF-74  
4. Safety Evaluation

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\*For previous concurrences see attached ORC

DOCUMENT NAME: PV94295.AMD

OFC	LA:PDIV-2	PDIV-2	PDIV-2	SCSB*	OGC*
NAME	EPeyton	CThomas	BHolian	CBerlinger	CMarco
DATE	2/12/96	2/13/96	2/21/96	1/30/96	2/13/96

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

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Mr. William L. Stewart

- 2 -

February 23, 1996

cc w/encls:

Mr. Steve Olea  
Arizona Corporation Commission  
1200 W. Washington Street  
Phoenix, Arizona 85007

T. E. Oubre, Esq.  
Southern California Edison Company  
P. O. Box 800  
Rosemead, California 91770

Senior Resident Inspector  
USNRC  
P. O. Box 40  
Buckeye, Arizona 85326

Regional Administrator, Region IV  
U. S. Nuclear Regulatory Commission  
Harris Tower & Pavillion  
611 Ryan Plaza Drive, Suite 400  
Arlington, Texas 76011-8064

Chairman, Board of Supervisors  
ATTN: Chairman  
301 W. Jefferson, 10th Floor  
Phoenix, Arizona 85003

Mr. Aubrey V. Godwin, Director  
Arizona Radiation Regulatory Agency  
4814 South 40 Street  
Phoenix, Arizona 85040

Mr. Curtis Hoskins  
Executive Vice President and  
Chief Operating Officer  
Palo Verde Services  
2025 N. 3rd Street, Suite 200  
Phoenix, Arizona 85004

Roy P. Lessey, Jr., Esq.  
Akin, Gump, Strauss, Hauer and Feld  
El Paso Electric Company  
1333 New Hampshire Avenue, Suite 400  
Washington, DC 20036

Ms. Angela K. Krainik, Manager  
Nuclear Licensing  
Arizona Public Service Company  
P.O. Box 52034  
Phoenix, Arizona 85072-2034



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. 103  
License No. NPF-41

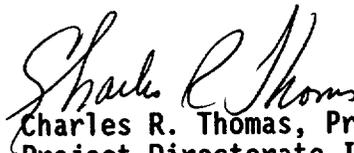
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated December 19, 1995, as supplemented by letter dated February 9, 1996, 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. 103, 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. The license amendment is effective as of its date of issuance to be implemented within 15 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Charles R. 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: February 23, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 103 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

1-2  
3/4 6-1  
3/4 6-2  
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B 3/4 6-1  
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INSERT

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## 1.0 DEFINITIONS

---

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

### ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

### AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

### AZIMUTHAL POWER TILT - $T_q$

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

### CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

### CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

## DEFINITIONS

### CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.
- c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.
- d. Radiological effluent process monitoring channels - the CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is functionally tested.

The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

### CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
  1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or
  2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.
- b. All equipment hatches are closed and sealed,
- c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

### CONTROLLED LEAKAGE

1.8 Not Applicable.

### CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

### 3/4.6 CONTAINMENT SYSTEMS

#### 3/4.6.1 PRIMARY CONTAINMENT

##### CONTAINMENT INTEGRITY

##### LIMITING CONDITION FOR OPERATION

---

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

##### SURVEILLANCE REQUIREMENTS

---

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

- a. At least once per 31 days by verifying that all penetrations\* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except for valves that are open under administrative control as permitted by Specification 3.6.3.
- b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

---

\*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

---

3.6.1.2 Containment leakage rates shall in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

With containment leakage rates not within limits, restore containment leakage rates to within limits within 1 hour, or be in at least HOT STANDBY within the next 6 hours, and at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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4.6.1.2 The containment leakage rates shall be determined in accordance with the Containment Leakage Rate Testing Program. The provisions of Specification 4.0.2 are not applicable.

## 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  $P_a$ , 49.5 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

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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.

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

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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- b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

## CONTAINMENT SYSTEMS

### INTERNAL PRESSURE

#### LIMITING CONDITION FOR OPERATION

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3.6.1.4 Primary containment internal pressure shall be maintained between -0.3 and 2.5 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

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4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

## CONTAINMENT SYSTEMS

### 3/4.6.3 CONTAINMENT ISOLATION VALVES

#### LIMITING CONDITION FOR OPERATION

---

3.6.3 Each containment isolation valve shall be OPERABLE\*

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

ACTION:

1. With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:
  - a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or
  - b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position\*\*, or
  - c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange\*\*; or
  - d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
  - e. The provisions of Specification 3.0.4 do not apply.

#### SURVEILLANCE REQUIREMENTS

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4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit.

4.6.3.2 Each isolation valve used in containment isolation, containment spray, or containment purge shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

- a. Verifying that on a CIAS, CSAS or SIAS test signal, each isolation valve actuates to its isolation position.
- b. Verifying that on a CPIAS test signal, all containment purge valves actuate to their isolation position.

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\*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

\*\*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

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4.6.3.3 The isolation time of each power operated or automatic valve used in CIAS, CPIAS, or CSAS shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The containment isolation check valves shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

4.6.3.5 The containment isolation valves used as safety/relief, normally open—ESF actuated closed, or required open during accident conditions shall be demonstrated OPERABLE as required by Specification 4.0.5 and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured\*\*\* in their actuated position are considered operable pursuant to this specification.

4.6.3.6 The manual containment isolation valves (normally closed/post accident closed valves) shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.

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\*Locked, sealed, or otherwise prevented from unintentional operation.

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 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  $0.75 L_t$  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

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#### 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 peak pressure 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 49.5 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 49.5 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 49.5 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.



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DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92  
License No. NPF-51

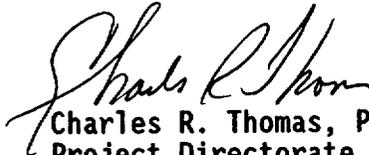
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated December 19, 1995, as supplemented by letter dated February 9, 1996, 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. 92, 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. The license amendment is effective as of its date of issuance to be implemented within 15 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Charles R. 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: February 23, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 92 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

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B 3/4 6-1  
6-25  
6-26

## 1.0 DEFINITIONS

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The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

### ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

### AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

### AZIMUTHAL POWER TILT - $T_g$

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

### CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

### CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

## DEFINITIONS

### CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.
- c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.
- d. Radiological effluent process monitoring channels - the CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is functionally tested.

The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

### CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
  1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or
  2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.
- b. All equipment hatches are closed and sealed,
- c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

### CONTROLLED LEAKAGE

1.8 Not Applicable.

### CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

### 3/4.6 CONTAINMENT SYSTEMS

#### 3/4.6.1 PRIMARY CONTAINMENT

##### CONTAINMENT INTEGRITY

##### LIMITING CONDITION FOR OPERATION

---

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

##### SURVEILLANCE REQUIREMENTS

---

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

- a. At least once per 31 days by verifying that all penetrations\* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except for valves that are open under administrative control as permitted by Specification 3.6.3.
- b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

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\*Except valves, blind flanges and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

## CONTAINMENT SYSTEMS

### CONTAINMENT LEAKAGE

#### LIMITING CONDITION FOR OPERATION

---

3.6.1.2 Containment leakage rates shall be in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

With containment leakage rates not within limits, restore containment leakage rates within limits within 1 hour, or be in at least HOT STANDBY within the next 6 hours, and at least COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

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4.6.1.2 The containment leakage rates shall be determined in accordance with the Containment Leakage Rate Testing Program. The provisions of Specification 4.0.2 are not applicable.

## 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  $P_a$ , 49.5 psig.

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

#### ACTION:

- a. With one containment air lock door inoperable:
  1. Maintain at least the OPERABLE airlock 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

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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.

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

**CONTAINMENT SYSTEMS**

**SURVEILLANCE REQUIREMENTS (Continued)**

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- b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

## CONTAINMENT SYSTEMS

### INTERNAL PRESSURE

#### LIMITING CONDITION FOR OPERATION

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3.6.1.4 Primary containment internal pressure shall be maintained between -0.3 and 2.5 psig.

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

#### ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

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4.6.1.4 The primary containment internal pressure shall be determined to be within the Limits at least once per 12 hours.

## CONTAINMENT SYSTEMS

### 3/4.6.3 CONTAINMENT ISOLATION VALVES

#### LIMITING CONDITION FOR OPERATION

---

3.6.3 Each containment isolation valve shall be OPERABLE.\*

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

#### ACTION:

1. With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:
  - a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or
  - b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position\*\*, or
  - c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange\*\*; or
  - d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
  - e. The provisions of Specification 3.0.4 do not apply.

#### SURVEILLANCE REQUIREMENTS

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4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit.

4.6.3.2 Each isolation valve used in containment isolation, containment spray, or containment purge shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

- a. Verifying that on a CIAS, CSAS or SIAS test signal, each isolation valve actuates to its isolation position.
- b. Verifying that on a CPIAS test signal, all containment purge valves actuate to their isolation position.

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\*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

\*\*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

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4.6.3.3 The isolation time of each power operated or automatic valve used in CIAS, CPIAS, or CSAS shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The containment isolation check valves shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

4.6.3.5 The containment isolation valves used as safety/relief, normally open—ESF actuated closed, or required open during accident conditions shall be demonstrated OPERABLE as required by Specification 4.0.5 and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured\*\*\* in their actuated position are considered operable pursuant to this specification.

4.6.3.6 The manual containment isolation valves (normally closed/post accident closed valves) shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.

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\*\*\*Locked, sealed, or otherwise prevented from unintentional operation.

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 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,  $P_a$ . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to  $0.75 L_p$  or less than or equal to  $0.75 L_c$ , 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

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#### 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 peak pressure 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 49.5 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 49.5 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 49.5 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

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6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM (Continued)

- 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 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75  
License No. NPF-74

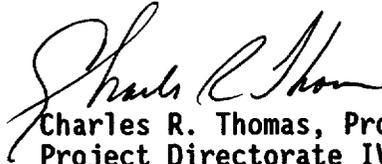
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated December 19, 1995, as supplemented by letter dated February 9, 1996, 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. 75, 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. The license amendment is effective as of its date of issuance to be implemented within 15 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Charles R. 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: February 23, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 75 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.

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## 1.0 DEFINITIONS

---

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

### ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

### AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

### AZIMUTHAL POWER TILT - $T_g$

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

### CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

### CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

## DEFINITIONS

### CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.
- c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.
- d. Radiological effluent process monitoring channels - the CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is functionally tested.

The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

### CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
  1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or
  2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.
- b. All equipment hatches are closed and sealed,
- c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellow or O-rings) is OPERABLE.

### CONTROLLED LEAKAGE

1.8 Not Applicable.

### CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

### 3/4.6 CONTAINMENT SYSTEMS

#### 3/4.6.1 PRIMARY CONTAINMENT

##### CONTAINMENT INTEGRITY

##### LIMITING CONDITION FOR OPERATION

---

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

##### SURVEILLANCE REQUIREMENTS

---

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

- a. At least once per 31 days by verifying that all penetrations\* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except for valves that are open under administrative control as permitted by Specification 3.6.3.
- b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

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\*Except valves, blind flanges and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

## CONTAINMENT SYSTEMS

### CONTAINMENT LEAKAGE

#### LIMITING CONDITION FOR OPERATION

---

3.6.1.2 Containment leakage rates shall be in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

With containment leakage rates not within limits, restore containment leakage rates to within limits within 1 hour or be in at least HOT STANDBY within the next 6 hours, and at least COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

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4.6.1.2 The containment leakage rates shall be determined in accordance with the Containment Leakage Rate Testing Program. The provisions of Specification 4.0.2 are not applicable.

## 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  $0.05 L_0$  at  $P_0$ , 49.5 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.

**CONTAINMENT SYSTEMS**

**SURVEILLANCE REQUIREMENTS (Continued)**

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- b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

## CONTAINMENT SYSTEMS

### INTERNAL PRESSURE

#### LIMITING CONDITION FOR OPERATION

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3.6.1.4 Primary containment internal pressure shall be maintained between -0.3 and 2.5 psig.

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

#### ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

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4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

## CONTAINMENT SYSTEMS

### 3/4.6.3 CONTAINMENT ISOLATION VALVES

#### LIMITING CONDITION FOR OPERATION

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3.6.3 Each containment isolation valve shall be OPERABLE\*

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

ACTION:

1. With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:
  - a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or
  - b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position\*<sup>†</sup>; or
  - c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange\*\*<sup>†</sup>; or
  - d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
  - e. The provisions of Specification 3.0.4 do not apply.

#### SURVEILLANCE REQUIREMENTS

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4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit.

4.6.3.2 Each isolation valve used in containment isolation, containment spray, or containment purge shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

- a. Verifying that on a CIAS, CSAS or SIAS test signal, each isolation valve actuates to its isolation position.
- b. Verifying that on a CPIAS test signal, all containment purge valves actuate to their isolation position.

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\*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

\*\*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

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4.6.3.3 The isolation time of each power operated or automatic valve used in CIAS, CPIAS, or CSAS shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The containment isolation check valves shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

4.6.3.5 The containment isolation valves used as safety/relief, normally open—ESF actuated closed, or required open during accident conditions shall be demonstrated OPERABLE as required by Specification 4.0.5 and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured\*\*\* in their actuated position are considered operable pursuant to this specification.

4.6.3.6 The manual containment isolation valves (normally closed/post accident closed valves) shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.

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\*Locked, sealed, or otherwise prevented from unintentional operation.

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 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,  $P_a$ . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to  $0.75 L_p$  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

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#### 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 peak pressure 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 49.5 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 49.5 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-6.8-1994.

The peak calculated containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 49.5 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

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### 6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM (Continued)

- 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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-41,  
AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-51,  
AND AMENDMENT NO. 75 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 19, 1995, 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 the technical specifications (TSS) to reflect the approval for the licensee to use 10 CFR Part 50, Appendix J, Option B for the Palo Verde Nuclear Generating Station containment leakage rate test program.

As reflected in a letter dated February 9, 1996, modifications were made to the technical specifications to be consistent with Appendix J, Option B. The February 9, 1996, letter did not change the staff's proposed determination that no significant hazards consideration were present.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate values specified in the Technical Specifications and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," to Part 50 of Title 10 of the Code of Federal Regulations was considered for this initiative, and the staff undertook a study of possible changes to this

regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program".

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," was developed as a method acceptable to the staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the staff for complying with Option B with four exceptions which are described therein.

Option B requires that Regulatory Guide 1.163 or another implementation document used by a licensee to develop a performance based leakage rate testing program must be included, by general reference, in the plant technical specifications.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum of 10 years based upon completion of two consecutive successful tests, and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed technical specifications for implementing Option B. After some discussion, the staff and NEI agreed on a set of model technical specifications, which were transmitted to NEI in a letter dated November 2, 1995. These technical specifications are to serve as a model for licensees to develop plant-specific technical specifications in preparing an amendment request to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to ensure that they are selected in a reasonable manner, they are not technical specification requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

### 3.0 EVALUATION

By letter dated December 19, 1995, Arizona Public Service Company proposed to establish a "Primary Containment Leakage Rate Program" and proposed to add this program to the technical specifications. The program references Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the staff for complying with Option B. This requires a change to existing Technical Specifications 3/4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3, and the addition of the program to Section 6.16 of the technical specifications.

Option B permits a licensee to do Type A testing, or Type B and C testing, or Type A, B, and C testing on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The technical specifications changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of Regulatory Guide 1.163, dated September 1995, and the generic technical specifications of the November 2, 1995 letter and are therefore acceptable to the staff.

### 4.0 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.

### 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 (61 FR 1627). 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 amendment.

**6.0 CONCLUSION**

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

**Principal Contributor: C. Thomas**

**Date: February 23, 1996**