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

May 16, 1994

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

Mr. William F. Conway
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

Dear Mr. Conway:

SUBJECT: ISSUANCE OF AMENDMENTS FOR THE PALO VERDE NUCLEAR GENERATING STATION UNIT NO. 1 (TAC NO. M79226), UNIT NO. 2 (TAC NO. M79227), AND UNIT NO. 3 (TAC NO. M79228)

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. NPF-41, Amendment No. 61 to Facility Operating License No. NPF-51, and Amendment No. 47 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 November 13, 1990, as supplemented by letters of additional information dated May 27, 1992, May 13, 1993, and November 12, 1993.

These amendments involve increasing the pressurizer safety valve (PSV) setpoint tolerance from +/-1 percent to +3 percent and -1 percent, the main steam safety valve (MSSV) setpoint tolerance from +/-1 percent to +/-3 percent, reducing the high pressurizer pressure trip setpoint (HPPT) response time from 1.15 seconds to 0.5 second, and reducing the TS minimum auxiliary feedwater (AFW) pump flow requirement from 750 gallons per minute (GPM) to 650 GPM.

A copy of the related Safety Evaluation is also enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

B E. Hol

Brian E. Holian, Project Manager
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 75 to NPF-41
2. Amendment No. 61 to NPF-51
3. Amendment No. 47 to NPF-74
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

Mr. William F. Conway
Arizona Public Service Company

Palo Verde

cc:

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P. O. Box 52034
Phoenix, Arizona 85072-2034

A copy of the related Safety Evaluation is also enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by:

Brian E. Holian, Project Manager
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 75 to NPF-41
2. Amendment No. 61 to NPF-51
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cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	KEccleston
NRC & Local PDRs	GHill (6), P1-37
PDIV-3 R/F	CGrimes, 11E22
JRoe	OPA, 2G5
EAdensam	CLiang
TQuay	GHammer
LTran	OC/LFDCB, 4503
TPolich	OGC, 15B18
DFoster-Curseen	KPerkins, RIV/WCFO
DHagan, 3206	RIV/WCFO (12)
ACRS (10), P-315	BHolian

* See previous concurrence

OFFICE	PDIV-3/LA	PDIV-3/PM	PDIV-3/PM*	SRXB*	EMEB*
NAME	DFoster-Curseen	LTran	BHolian:mk	TCollins	JNorberg
DATE	5/5/94	5/3/94	3/30/94	3/08/94	3/10/94

OFFICE	PRPB*	OGC*	PDIV-3/D
NAME	LJCunningham	CBarth	TQuay
DATE	3/15/94	02/03/94	5/16/94

OFFICIAL RECORD COPY DOCUMENT NAME: PV79226.AMD

A copy of the related Safety Evaluation is also enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by:

Brian E. Holian, Project Manager
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 75 to NPF-41
2. Amendment No. 61 to NPF-51
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ACRS (10), P-315	BHolian

* See previous concurrence

OFFICE	PDIV-3/LA	PDIV-3/PM	PDIV-3/PM*	SRXB*	EMEB*
NAME	DFoster-Curseen	LTran	BHolian:mk	TCollins	JNorberg
DATE	5/5/94	5/3/94	3/30/94	3/08/94	3/10/94

OFFICE	PRPB*	OGC*	PDIV-3/D
NAME	LJCunningham	CBarth	TQuay
DATE	3/15/94	02/03/94	5/16/94

OFFICIAL RECORD COPY DOCUMENT NAME: PV79226.AMD

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.75
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 November 13, 1990, supplemented by letters dated May 27, 1992, May 13, 1993, and November 12, 1993, 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:

3PP

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(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. This license amendment is effective as of the date of issuance and must be fully implemented no later than December 1, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 16, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 75 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 vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
3/4 3-11	3/4 3-11
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 7-2	3/4 7-2
3/4 7-5	3/4 7-5
B 3/4 7-2	B 3/4 7-2

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 0.50 seconds
2. Pressurizer Pressure Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ 0.58 second
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. Cold Leg Temperature	≤ 0.75 second##
d. Hot Leg Temperature	≤ 0.75 second##
e. Primary Coolant Pump Shaft Speed	≤ 0.30 second#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.75 second###
g. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 0.55 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	$\leq$ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

* Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

** Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 8 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia +3, -1%*.

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia +3, -1%*.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

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

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1 and 2 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the maximum variable overpower trip setpoint and the maximum Allowable Steady State Power Level are reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with at least one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

VALVE NUMBER			LIFT SETTING (±3%)*	MINIMUM RATED CAPACITY**
	S/G No. 1	S/G No. 2		
a.	SGE PSV 572	SGE PSV 554	1250 psig	941,543 lb/hr
b.	SGE PSV 579	SGE PSV 561	1250 psig	941,543 lb/hr
c.	SGE PSV 573	SGE PSV 555	1290 psig	971,332 lb/hr
d.	SGE PSV 578	SGE PSV 560	1290 psig	971,332 lb/hr
e.	SGE PSV 574	SGE PSV 556	1315 psig	989,950 lb/hr
f.	SGE PSV 575	SGE PSV 557	1315 psig	989,950 lb/hr
g.	SGE PSV 576	SGE PSV 558	1315 psig	989,950 lb/hr
h.	SGE PSV 577	SGE PSV 559	1315 psig	989,950 lb/hr
i.	SGE PSV 691	SGE PSV 694	1315 psig	989,950 lb/hr
j.	SGE PSV 692	SGE PSV 695	1315 psig	989,950 lb/hr

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation.

SURVEILLANCE REQUIREMENTS

b. At least once per 18 months during shutdown by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

   2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying on a STAGGERED TEST BASIS (by means of a flow test) that the normal flow path from the condensate storage tank to each of the steam generators through one of the essential auxiliary feedwater pumps delivers at least 650 gpm at 1270 psia or equivalent at the entrance of the steam generator.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 for the turbine-driven pump.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with an indicated level of at least 25 feet (300,000 gallons).

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

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the essential auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the essential auxiliary feedwater pumps by verifying:

a. That the reactor makeup water tank supply line to the auxiliary feedwater system isolation valve is open, and

b. That the reactor makeup water tank contains a water level of at least 26 feet (300,000 gallons).

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*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes the valves are sized to pass a minimum of 102% of the RATED THERMAL POWER at 102% of design power. The adequacy of this relieving capacity is demonstrated by maintaining the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks, CEA ejection and 110% of design pressure for all overpressurization events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition including the Summer 1975 Addenda. The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is 19.44×10^6 lbm/hr. This capacity is less than the total rated capacity as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainty) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83×10^6 lbm/hr. The ratio of this total steam flow to the total capacity is 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of the total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

The ceiling on the variable over power reactor trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

$$SP = \text{Allowable Power Level} + 9.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

SAFETY VALVES (continued)

10	=	total number of secondary safety valves for one steam generator.
N	=	number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves.
109.2	=	ratio of main steam safety valve relieving capacity of 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainty (see above text)
9.8	=	BAND between the maximum thermal power and the variable over power trip setpoint ceiling

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 650 gpm at a pressure of 1270 psia at the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 650 gpm at a pressure of 1270 psia at the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ensures that a minimum water volume of 300,000 gallons is available to maintain the Reactor Coolant System at HOT STANDBY for 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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. 61
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 November 13,1990, supplemented by letters dated May 27, 1992, May 13, 1993, and November 12, 1993, 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 Part 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. 61, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and must be fully implemented no later than December 1, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 16, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 61 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 vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
3/4 3-11	3/4 3-11
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 7-2	3/4 7-2
3/4 7-5	3/4 7-5
B 3/4 7-2	B 3/4 7-2

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 0.50 seconds
2. Pressurizer Pressure Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ 0.58 second
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. Cold Leg Temperature	≤ 0.75 second##
d. Hot Leg Temperature	≤ 0.75 second##
e. Primary Coolant Pump Shaft Speed	≤ 0.30 second#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.75 second###
g. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 0.55 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	≤ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

---

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 8 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia +3, -1%*.

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia +3, -1%*.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

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

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1 and 2 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Maximum Variable Overpower trip setpoint and the Maximum Allowable Steady State Power Level are reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with at least one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

VALVE NUMBER			LIFT SETTING (±3%) *	MINIMUM RATED CAPACITY**
	S/G No. 1	S/G No. 2		
a.	SGE PSV 572	SGE PSV 554	1250 psig	941,543 lb/hr
b.	SGE PSV 579	SGE PSV 561	1250 psig	941,543 lb/hr
c.	SGE PSV 573	SGE PSV 555	1290 psig	971,332 lb/hr
d.	SGE PSV 578	SGE PSV 560	1290 psig	971,332 lb/hr
e.	SGE PSV 574	SGE PSV 556	1315 psig	989,950 lb/hr
f.	SGE PSV 575	SGE PSV 557	1315 psig	989,950 lb/hr
g.	SGE PSV 576	SGE PSV 558	1315 psig	989,950 lb/hr
h.	SGE PSV 577	SGE PSV 559	1315 psig	989,950 lb/hr
i.	SGE PSV 691	SGE PSV 694	1315 psig	989,950 lb/hr
j.	SGE PSV 692	SGE PSV 695	1315 psig	989,950 lb/hr

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation.

SURVEILLANCE REQUIREMENTS

b. At least once per 18 months during shutdown by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.*

   2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying on a STAGGERED TEST BASIS (by means of a flow test) that the normal flow path from the condensate storage tank to each of the steam generators through one of the essential auxiliary feedwater pumps delivers at least 650 gpm at 1270 psia or equivalent at the entrance of the steam generator.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 for the turbine-driven pump.

*Deferred until cycle 3 refueling outage.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with an indicated level of at least 25 feet (300,000 gallons).

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

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the auxiliary feedwater pumps by verifying:

a. That the reactor makeup water tank supply line to the auxiliary feed system isolation valve is open, and

b. That the reactor makeup water tank contains a water level of at least 26 feet (300,000 gallons).

*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes the valves are sized to pass a minimum of 102% of the RATED THERMAL POWER at 102% of design power. The adequacy of this relieving capacity is demonstrated by maintaining the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks, CEA ejection and 110% of design pressure for all overpressurization events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition including the Summer 1975 Addenda. The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is 19.44×10^6 lbm/hr. This capacity is less than the total rated capacity as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainty) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83×10^6 lbm/hr. The ratio of this total steam flow to the total capacity is 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of the total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

The ceiling on the variable over power reactor trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

$$SP = \text{Allowable Power Level} + 9.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

SAFETY VALVES (continued)

10	=	total number of secondary safety valves for one steam generator.
N	=	number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves.
109.2	=	ratio of main steam safety valve relieving capacity of 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainty (see above text)
9.8	=	BAND between the maximum thermal power and the variable over power trip setpoint ceiling

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 650 gpm at a pressure of 1270 psia at the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 650 gpm at a pressure of 1270 psia at the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ensures that a minimum water volume of 300,000 gallons is available to maintain the Reactor Coolant System at HOT STANDBY for 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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.47
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 November 13, 1990, supplemented by letters dated May 27, 1992, May 13, 1993, and November 12, 1993, 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. 47 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and must be fully implemented no later than December 1, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 16, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 47 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 vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
3/4 3-11	3/4 3-11
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 7-2	3/4 7-2
3/4 7-5	3/4 7-5
B 3/4 7-2	B 3/4 7-2

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 0.50 seconds
2. Pressurizer Pressure Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ 0.58 second
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. Cold Leg Temperature	≤ 0.75 second##
d. Hot Leg Temperature	≤ 0.75 second##
e. Primary Coolant Pump Shaft Speed	≤ 0.30 second#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.75 second###
g. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 0.55 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	$\leq$ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 8 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia +3, -1%*.

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia +3, -1%*.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

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

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1 and 2 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Variable Overpower trip setpoint ceiling and the Maximum Allowable Steady State Power Level are reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with at least one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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*Until the steam generators are no longer required for heat removal.
**The maximum number of inoperable safety valves on any operating steam generator is four (4).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

VALVE NUMBER	S/G No. 1	S/G No. 2	LIFT SETTING (±3%) *	MINIMUM RATED CAPACITY**
a.	SGE PSV 572	SGE PSV 554	1250 psig	941,543 lb/hr
b.	SGE PSV 579	SGE PSV 561	1250 psig	941,543 lb/hr
c.	SGE PSV 573	SGE PSV 555	1290 psig	971,332 lb/hr
d.	SGE PSV 578	SGE PSV 560	1290 psig	971,332 lb/hr
e.	SGE PSV 574	SGE PSV 556	1315 psig	989,950 lb/hr
f.	SGE PSV 575	SGE PSV 557	1315 psig	989,950 lb/hr
g.	SGE PSV 576	SGE PSV 558	1315 psig	989,950 lb/hr
h.	SGE PSV 577	SGE PSV 559	1315 psig	989,950 lb/hr
i.	SGE PSV 691	SGE PSV 694	1315 psig	989,950 lb/hr
j.	SGE PSV 692	SGE PSV 695	1315 psig	989,950 lb/hr

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation.

SURVEILLANCE REQUIREMENTS

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying on a STAGGERED TEST BASIS (by means of a flow test) that the normal flow path from the condensate storage tank to each of the steam generators through one of the essential auxiliary feedwater pumps delivers at least 650 gpm at 1270 psia or equivalent at the entrance of the steam generator.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 for the turbine-driven pump.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least 25 feet (300,000 gallons).

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

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the essential auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the essential auxiliary feedwater pumps by verifying:

a. That the reactor makeup water tank supply line to the auxiliary feedwater system isolation valve is open, and

b. That the reactor makeup water tank contains a water level of at least 26 feet (300,000 gallons).

*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes the valves are sized to pass a minimum of 102% of the RATED THERMAL POWER at 102% of design power. The adequacy of this relieving capacity is demonstrated by maintaining the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks, CEA ejection and 110% of design pressure for all overpressurization events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition including the Summer 1975 Addenda. The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is 19.44×10^6 lbm/hr. This capacity is less than the total rated capacity as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainty) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is 17.83×10^6 lbm/hr. The ratio of this total steam flow to the total capacity is 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of the total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

The ceiling on the variable over power reactor trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

$$SP = \text{Allowable Power Level} + 9.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

SAFETY VALVES (continued)

10 = total number of secondary safety valves for one steam generator.

N = number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves.

109.2 = ratio of main steam safety valve relieving capacity of 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainty (see above text)

9.8 = BAND between the maximum thermal power and the variable over power trip setpoint ceiling

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 650 gpm at a pressure of 1270 psia at the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 650 gpm at a pressure of 1270 psia at the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ensures that a minimum water volume of 300,000 gallons is available to maintain the Reactor Coolant System at HOT STANDBY for 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. 47 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 letter dated November 13, 1990, the Arizona Public Service Company (APS or the licensee) submitted a request for changes to the Technical Specifications (TS) for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (Appendix A to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, respectively). The 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 involve increasing the pressurizer safety valve (PSV) setpoint tolerance from +/-1 percent to +3 percent and -1 percent, the main steam safety valve (MSSV) setpoint tolerance from +/-1 percent to +/-3 percent, reducing the high pressurizer pressure trip setpoint (HPPT) response time from 1.15 seconds to 0.5 second, and reducing the TS minimum auxiliary feedwater (AFW) pump flow requirement from 750 gallons per minute (GPM) to 650 GPM. The TS PSV setpoint tolerance is proposed to be increased to +3 percent; however, the -1 percent tolerance is necessary to ensure that the reactor is tripped before the PSVs open. The licensee also submitted a letter dated April 29, 1991, which committed to reset the setpoint of PSVs and MSSVs found to have setpoints above or below the +/-1 percent range to within +/-1 percent of the nominal setpoint. In response to concerns identified by the staff, the licensee also submitted letters dated May 27, 1992, and May 13, 1993, and November 12, 1993, which provided additional information in support of the proposed changes to the PSV and MSSV setpoint tolerances. The proposed amendment affects TS Sections 3/4.3.1, 3/4.4.2, 3/4.7.1 and 3/4.7.1.2 for the PSV and MSSV tolerances, HPPT response time, and auxiliary feedwater flow.

During the review of this Technical Specification application, the staff became aware of a minor "offset" that exists between the two accepted testing methods for verifying valve setpoint. The "trevitest" method (stem pulled mechanically) was generally lower than the live steam method of testing. This

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discrepancy is apparently due to an estimation of the valve's mean seat area in the equation used to calculate the appropriate setpoint. The licensee identified this discrepancy and conservatively used the live steam method as a basis for setting the valve setpoints. Therefore, this issue of reconciling the test methods does not impact on the setpoint tolerance evaluation that follows. The staff is currently evaluating differences in the test methods and will address this issue separately.

2.0 EVALUATION

To support the proposed TS revisions for PSV and MSSV setpoint tolerances, HPPT response time, and AFW requirements, the licensee has performed an evaluation and analyses to determine the impact on the design basis transients and accidents for Palo Verde 1, 2, and 3. All of the transient and accident analyses documented in the UFSAR were evaluated by APSC to determine the impact of the proposed changes to the TS. For the cases where the TS changes had an adverse impact on event consequences, a detailed evaluation or reanalysis of the event has been performed. APSC has provided detailed evaluations for the following events: (1) loss of condenser vacuum (LOCV), (2) main feedwater line break (MFLB), (3) steam generator tube rupture (SGTR), (4) reactor coolant pump (RCP) shaft seizure, and (5) loss of coolant accident (LOCA). The staff approved CESEC III code was used in performing the APSC evaluation. The most conservative assumptions including the proposed changes to the TS were used to demonstrate that the acceptance criteria for each analyzed event (e.g. peak system pressure, fuel performance, and offsite dose) were met.

For events where the maximum system pressure was a concern, such as LOCV and MFLB, the PSVs and MSSVs were assumed to have setpoints at the maximum tolerance value of +3 percent. The PSVs have been assumed to fully open at these setpoints to immediately deliver full rated discharge flow, whereas in the current FSAR analyses, the valves were modelled to open only to 70 percent of the nominal area opening at the setpoint. The licensee based this new assumption on test results which show that the valves attain full lift in 0.02 seconds after reaching the setpoint. The licensee stated that the maximum additional accumulation in pressure for this .02 second delay in valve lifting would be about 2 psi more than the analysis result (still within the 110 percent of design pressure safety limit). These analyses also assume the proposed HPPT response time of 0.5 seconds and the proposed AFW flow of 650 GPM. The licensee stated that the reduced HPPT response time is supported by actual plant test results which show that the actual time is less than 0.3 seconds. This change would result in an early reactor trip following an analyzed heatup transient and would reduce the system peak pressure during the analyzed transient. The licensee stated that the 650 GPM of AFW flow is supported by the plant analyses which meet the necessary criteria for this minimum amount of flow. In the analyses, the licensee also reduced the surge line form loss factor to 3.0 from 3.9 which is based on a calculation for the actual plant configuration. The LOCV event is the limiting transient of moderate frequency occurrence for maximum overpressure. The maximum allowable primary system pressure is 2750 psia (110 percent of design pressure), and the maximum reanalyzed primary pressure for this event is 2740.9 psia. The maximum allowable secondary system pressure is 1375 psia (110 percent of design pressure), and the maximum reanalyzed secondary pressure for this event is 1369.6 psia. The licensee also provided the results of its reanalysis of the MFLB event which is the design basis event producing the maximum primary system pressure condition. The reanalysis of the MFLB resulted in a peak RCS pressure of 2816 psia which is less than the peak pressure of 2843 psia documented in the UFSAR. The results of the APS reanalyses of the SGTR, RCP shaft seizure, and LOCA which consider the proposed TS changes also demonstrate that the acceptance criteria for these events are met as provided in the Standard Review Plan.

In the review of the licensee's submittals, one of the staff concerns relates to the potential overall reduction of conservatism in order to meet the required limits for system overpressurization and other acceptance criteria, especially for the LOCV event. The licensee has provided the following information regarding the conservatisms which exist in the analysis to demonstrate that the maximum allowable pressure will not be exceeded:

1. Feedwater and steam flow actually ramp down to zero in about 18 seconds instead of the analyzed 0.1 second.

2. Safety valves are assumed to open at the +3 percent setpoint tolerance, whereas some are actually expected to open at lower pressures.

3. There is 30 psi of additional conservatism in the high pressurizer/pressure trip setpoint of 2540 psia. Also, surveillance tests indicate that this trip response time is less than 0.3 second instead of the assumed 0.5 second.

4. The analysis does not assume that the pressurizer spray valves open.

5. The initial pressurizer level in the analysis is conservative compared to the level normally expected.

6. Non-safety systems, such as the Reactor Power Cutback System and the Steam Bypass Control System, are assumed to not operate in the analysis.

7. The moderator temperature coefficient is assumed most positive in the analysis.

8. Other conservative conditions regarding the reactor physics parameters in the analysis are: the least negative fuel temperature coefficient is assumed, bounding generic kinetic parameters are used, and the most limiting control rod is assumed to be stuck full out.

The licensee stated that with the operating conditions experienced most of the time, the peak pressure for the LOCV would be only 2650.5 psia. The licensee also states that the ASME Code provides assurance of large margin to failure and that the analysis performed to support the TS changes is adequate. The staff determined that if the analysis demonstrates the adequacy of the system overpressure protection with sufficiently conservative deterministic criteria and acceptably conservative input parameters and analysis methodology, then no minimal margin beyond the acceptance criteria is required. The staff has reviewed the licensee's analysis methodology and input assumptions and agrees that they are sufficiently conservative. Thus the licensee has met the required limits for overpressurization and other acceptance criteria, and demonstrated that maximum allowable pressure would not be exceeded.

Regarding the specific analysis model of PSV performance, the staff agrees that the modelling of the PSVs to open fully at their setpoints (with the +3 percent tolerance) is acceptable based on PSV test data. The previous method used in modelling the PSV performance involved opening the valve to only 70 percent open at the setpoint pressure. The proposed method is more nearly a best-estimate modelling technique (i.e. within 2 psi of actual expected performance as discussed above,) and as such provides no additional conservatism beyond that required to meet the acceptance criterion. Although the previous method is more conservative than that being currently proposed, the staff agrees that the overall conservatism of the analysis assumptions taken together is adequate.

Another of the staff concerns relates to the proper actions the licensee should pursue to assure the best performance of the plant PSVs and MSSVs. Plant operating history has shown that these valves frequently have not met the +/-1 percent setpoint tolerance criterion, and some also have not met the +/-3 percent criterion. Since there is little additional margin in the overpressure analyses discussed above, additional setpoint drift beyond the +/-3 percent tolerance on all valves could result in exceeding the allowable limits in the analyses. The licensee has enhanced the maintenance activities for the PSVs and the MSSVs to both increase the number of valves tested and improve the testing methods over a sufficient period of time to determine the necessary actions to improve setpoint performance. In addition, the licensee is investigating the root cause of the past setpoint performance. The licensee plans to continue the enhanced maintenance and testing over a period of several testing cycles in order to verify improved valve performance. The licensee's activities include better testing procedures and acceptance criteria and provide for better control of the testing parameters and the valve adjustments. The staff agrees that the licensee's activities for identifying corrective actions for maintaining and testing the PSVs and MSSVs should improve the performance of these valves and minimize setpoint drift beyond the +/-3 percent tolerance.

Another of the staff concerns relates to the effect which the increased PSV setpoint range has on the analyzed maximum pressurizer level. If the pressurizer should fill and subcooled liquid is discharged through the PSVs, the valves may not perform adequately. The full-flow tests performed on full scale PSV models by the Electric Power Research Institute (EPRI) in 1981 demonstrated unreliable performance when discharging some subcooled liquid conditions. The licensee performed an analysis to demonstrate that the pressurizer will not fill for the limiting transient (LOCV event) and that the PSVs will discharge only steam. The analysis used initial conditions set to maximize the pressurizer liquid level and the PSV blowdown was modelled to be 20 percent, which is the maximum observed in the EPRI tests. The staff agrees that the licensee's analysis is sufficient to demonstrate that the pressurizer will not fill for the limiting event and that the PSVs will discharge only steam.

The staff also reviewed the licensee's assessment of increased radiological release as a result of the safety valve set point tolerance change and the proposed reduction in auxiliary feedwater flow. A steam generator mass balance was performed by the licensee that showed an increase in the duration of tube uncovery from about 14.8 minutes to 18.3 minutes. The offsite radiological releases for a SGTR are strongly dependent on the partitioning of the primary to secondary leakage (the licensee's UFSAR assumes a maximum partition coefficient of less than 0.1 with tubes covered; and 1.0, with tubes uncovered). Since a significant portion of the offsite radiological releases occur during the period of tube uncovery, the licensee conservatively postulated that offsite doses will also increase in proportion to the increase in the duration of the tube uncovery. For the most limiting SGTR scenario, this results in an increase in the 2-hour thyroid dose from 200 rem to less than 248 rem. Adding the increase in dose due to the expanded PSV and MSSV setpoint tolerances (which the licensee calculated as a 5 percent dose increase), results in a total 2-hour dose of 260 rem. This value provides adequate margin to the 10 CFR Part 100 guideline of 300 rem. The licensee also evaluated radiological release for the RCP shaft seizure event, and determined that the calculated .5 rem increase was insignificant compared to the 246 rem dose reported in Supplement 2 to the staff's Safety Evaluation Report related to the Combustion Engineering Standard Safety Analysis Report (CESSAR) for System 80, and the 300 rem SRP acceptance criteria. The staff has reviewed the licensee's assessment of increased dosage and finds the results, utilizing proper conservatisms, within 10 CFR guidelines, and therefore acceptable.

The licensee also changed the bases to Technical Specification 3/4.7.1.2 to properly reflect the change to the surveillance requirement acceptance criteria of TS 4.7.1.2.c, for minimum auxiliary feedwater flow. The bases section and the associated TS were also clarified to properly reference steam generator pressure at the entrance of the steam generators. These changes are considered to be of clarification in nature, and are considered acceptable.

Based on the above evaluation, the staff agrees that the analysis which the licensee has provided demonstrates the acceptability of the proposed TS changes. The proposed increase in the setpoint tolerances of the PSVs to +3 percent and -1 percent and of the MSSVs to +/-3 percent has been shown to be acceptable for meeting the plant design basis, and the licensee's actions to improve the maintenance and testing of these valves should minimize the occurrence of valve setpoints outside this range. The reduction of the HPPT response time to 0.5 seconds has been shown to be conservative based on actual plant test data. The reduction of the required AFW flow to 650 GPM in the T/S as well as the other changes stated above are assumed in the reanalyses of the affected transients and accidents for Palo Verde Units 1, 2, and 3; and are therefore, acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on April 12, 1994 (59 FR 17403). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Hammer
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Date: May 16, 1994

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 75 to Facility Operating License No. NPF-41, Amendment No. 61 to Facility Operating License No. NPF-51, and Amendment No. 47 to Facility Operating License No. NPF-74, issued to Arizona Public Service Company, 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 (licensees), which revised the Technical Specifications for operation of the Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3 located in Maricopa County, Arizona. The amendment is effective as of the date of issuance.

The amendment modified the Technical Specifications to increase the pressurizer safety valve (PSV) setpoint tolerance from +/-1 percent to +3 percent and -1 percent, the main steam safety valve (MSSV) setpoint tolerance from +/-1 percent to +/-3 percent, reducing the high pressurizer pressure trip setpoint (HPPT) response time from 1.15 seconds to 0.5 second, and reducing the TS minimum auxiliary feedwater (AFW) pump flow requirement from 750 gallons per minute (GPM) to 650 GPM.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the

Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on December 27, 1990 (55 FR 53220). A request for a hearing or petition for leave to intervene was filed following this notice. By Memorandum and Order dated May 9, 1991 (LBP-91-19), a petition for leave to intervene and a request for hearing filed by the intervenors was granted. However, the intervenors later withdrew that challenge. By Memorandum and Order dated September 30, 1991 (LBP-91-37A), the proceeding was terminated.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment (59 FR 17403).

For further details with respect to the action see (1) the application for amendment dated November 13, 1990, and supplemented May 27, 1992, May 13, 1993, and November 12, 1993, (2) Amendment No. 75 to License No. NPF-41, (3) Amendment No. 61 to License No. NPF-51, (4) Amendment No. 47 to License No. NPF-74, (4) the Commission's related Safety Evaluation, and (5) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC 20555, and at the local public

document room located at Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Dated at Rockville, Maryland, this 16th day of May 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

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Project Directorate IV-3
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Office of Nuclear Reactor Regulation

UNITED STATES NUCLEAR REGULATORY COMMISSION

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3

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The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the

OP-1

Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on December 27, 1990 (55 FR 53220). A request for a hearing or petition for leave to intervene was filed following this notice. By Memorandum and Order dated May 9, 1991 (LBP-91-19), a petition for leave to intervene and a request for hearing filed by the intervenors was granted. However, the intervenors later withdrew that challenge. By Memorandum and Order dated September 30, 1991 (LBP-91-37A), the proceeding was terminated.

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Dated at Rockville, Maryland, this 16thday of 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

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