October 4, 1995

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. M93260), UNIT NO. 2 (TAC NO. M93261), AND UNIT NO. 3 (TAC NO. M93262)

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-41, Amendment No. 89 to Facility Operating License No. NPF-51, and Amendment No. 72 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 August 3, 1995.

These amendments add the analytical method supplement entitled "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, dated May 1990, and its associated Nuclear Regulatory Commission Safety Evaluation Report, dated April 10, 1990, to the list of analytical methods in Section 6.9.1.10 of the technical specifications used to determine the PVNGS core operating limits.

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. 101 to NPF-41
2. Amendment No. 89 to NPF-51
3. Amendment No. 72 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

Docket File	KPerkins, WCFO
PUBLIC	~~XXXXXXXXX~~
PDIV-2 Reading	GHill (6), T5C3
EGA1	OC/LFDCB, T9E10
JRoe	OGC, 015B18
WBateman	OPA, 02G5
CGrimes, 011E22	ACRS (4), T2E26
BHolian	LHurley, RIV
CThomas	JBianchi, WCFO (2)
EPeyton	RJones
	RHuey, RIV

9510130396 951004
PDR ADOCK 05000528
P PDR

DOCUMENT NAME: PV93260.AMD

OFC	PDIV-2/LA	PDIV-2/PM	PDIV-2/PM	NRR:SRXB	OGC RMW No legal objection with comments*
NAME	EPeyton	CThomas	BHolian	RJones	RWeisman
DATE	8/31/95	8/31/95	9/12/95	9/11/95	9/19/95

OFFICIAL RECORD COPY

*Provided notice period expired on 9/29/95.

120065

DF01

October 4, 199

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. M93260), UNIT NO. 2 (TAC NO. M93261), AND UNIT NO. 3 (TAC NO. M93262)

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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. 101 to NPF-41
2. Amendment No. 89 to NPF-51
3. Amendment No. 72 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION
Docket File
PUBLIC
PDIV-2 Reading
EGA1
JRoe
WBateman
CGrimes, O11E22
BHolian
CThomas
EPeyton
KPerkins, WCFO
~~[illegible]~~
GHill (6), T5C3
~~OC/LFDCB, T9E10~~
OGC, O15B18
~~OPA, O2G5~~
ACRS ~~(4)~~, T2E26
LHurley, RIV
JBianchi, WCFO (2)
RJones
RHuey, RIV

DOCUMENT NAME: PV93260.AMD

OFC	PDIV-2/LA	PDIV-2/PM	PDIV-2/PM	NRR:SRXB	OGC RMW No legal objection with comments*
NAME	EPeyton	CThomas	BHolian	RJones	RWeisman
DATE	8/31/95	8/31/95	10/2/95	9/11/95	9/19/95

OFFICIAL RECORD COPY

*Provided notice period expired on 9/29/95.

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

October 4, 1995

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. M93260), UNIT NO. 2 (TAC NO. M93261), AND UNIT NO. 3 (TAC NO. M93262)

Dear Mr. Stewart:

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These amendments add the analytical method supplement entitled "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, dated May 1990, and its associated Nuclear Regulatory Commission Safety Evaluation Report, dated April 10, 1990, to the list of analytical methods in Section 6.9.1.10 of the technical specifications used to determine the PVNGS core operating limits.

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Sincerely,

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. 101 to NPF-41
2. Amendment No. 89 to NPF-51
3. Amendment No. 72 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

cc w/encls:
Mr. Steve Olea
Arizona Corporation Commission
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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
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Phoenix, Arizona 85040

Mr. Curtis Hoskins
Executive Vice President and
Chief Operating Officer
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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. 101
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 August 3, 1995, 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 101, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance to be implemented prior to startup from RF06.

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: October 4, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-41

DOCKET NO. STN 50-528

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

REMOVE	INSERT
6-20b	6-20b

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin K_{N-1} - Any CEA Withdrawn for Specification 3.1.1.2
b. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
c. Boron Dilution Alarms for Specification 3.1.2.7
d. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
e. Regulating CEA Insertion Limits for Specification 3.1.3.6
f. Part Length CEA Insertion Limits for Specification 3.1.3.7
g. Linear Heat Rate for Specification 3.2.1
h. Azimuthal Power Tilt - T_q for Specification 3.2.3
i. DNBR Margin for Specification 3.2.4
j. Axial Shape Index for Specification 3.2.7

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis,"CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.2, Shutdown Margin K_{N-1} - Any CEA Withdrawn; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits and 3.1.3.6, Regulating CEA Insertion Limits).

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (Novenber 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin K_{N-1} - Any CEA Withdrawn; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

j. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

k. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 6.9.1.10.j.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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. 89
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 August 3, 1995, 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. 89, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance to be implemented prior to startup from RF06.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas

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: October 4, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. NPF-51

DOCKET NO. STN 50-529

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

REMOVE	INSERT
6-20b	6-20b

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin K_{N-1} - Any CEA Withdrawn for Specification 3.1.1.2
b. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
c. Boron Dilution Alarms for Specification 3.1.2.7
d. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
e. Regulating CEA Insertion Limits for Specification 3.1.3.6
f. Part Length CEA Insertion Limits for Specification 3.1.3.7
g. Linear Heat Rate for Specification 3.2.1
h. Azimuthal Power Tilt - T_q for Specification 3.2.3
i. DNBR Margin for Specification 3.2.4
j. Axial Shape Index for Specification 3.2.7

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.2, Shutdown Margin K_{N-1} - Any CEA Withdrawn; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits and 3.1.3.6, Regulating CEA Insertion Limits).

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (Novenber 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin K_{N-1} - Any CEA Withdrawn; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

j. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

k. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 6.9.1.10.j.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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. 72
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 August 3, 1995, 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. 72, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance to be implemented prior to startup from RF05.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas

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: October 4, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-74

DOCKET NO. STN 50-530

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

REMOVE	INSERT
6-20b	6-20b

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin K_{N-1} - Any CEA Withdrawn for Specification 3.1.1.2
b. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
c. Boron Dilution Alarms for Specification 3.1.2.7
d. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
e. Regulating CEA Insertion Limits for Specification 3.1.3.6
f. Part Length CEA Insertion Limits for Specification 3.1.3.7
g. Linear Heat Rate for Specification 3.2.1
h. Azimuthal Power Tilt - T_q for Specification 3.2.3
i. DNBR Margin for Specification 3.2.4
j. Axial Shape Index for Specification 3.2.7

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 (Methodology for Specifications 3.1.1.2, Shutdown Margin K_{N-1} - Any CEA Withdrawn; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits and 3.1.3.6, Regulating CEA Insertion Limits).

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (Novenber 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin K_{N-1} - Any CEA Withdrawn; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

j. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

k. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 6.9.1.10.j.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. 72 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 August 3, 1995, the Arizona Public Service Company (APS or the licensee) requested changes to the Technical Specifications (TS) (Appendix A to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74) for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, respectively. The licensee 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 amendments add the analytical method supplement entitled "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, dated May 1990, and its associated Nuclear Regulatory Commission Safety Evaluation Report, dated April 10, 1990, to the list of analytical methods in Section 6.9.1.10 of the TS, used to determine the PVNGS core operating limits.

2.0 BACKGROUND

The reload analyses for Unit 3 Cycle 6 indicate that internal fuel rod pressure will exceed reactor coolant system (RCS) pressure during Cycle 6. Internal fuel rod pressure increased because of many factors, such as burnup time for fuel assemblies, utilization of low leakage cores, mechanical modifications to fuel pellets and rod configuration, and the use of Erbia as a burnable absorber.

Asea Brown-Boveri-Combustion Engineering (ABB-CE) Topical Report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," provides technical justification for rod internal pressure exceeding RCS pressure. In its safety evaluation (SE) dated April 10, 1990, the NRC approved licensees' use of this methodology. However, licensees referencing the topical report are required to (1) prepare a plant-specific loss-of-coolant accident (LOCA) analysis to determine the impact of maximum calculated rod pressures on cladding rupture timing and peak cladding temperatures and (2) analyze departure from nucleate boiling (DNB) propagation in postulated accidents if the bounding 14 x 14 steamline break is not applicable for calculating maximum cladding rupture strain and percent flow blockage.

ABB-CE has completed the LOCA analysis for Unit 3 to determine the impact of maximum calculated rod pressures on cladding rupture timing and peak cladding temperature. APS requested that this proposed change to modify the list of analytical methods used for determining core operating limits be approved for all three units at PVNGS on the basis of the Unit 3 analysis methodology and results which would be similar for all three units. Because the three units at PVNGS are of identical design and construction, the staff has concluded that the analysis in this safety evaluation bounds all three units. Actual analyses for future PVNGS cycles will be performed as part of reload analyses as required. ABB-CE performed the PVNGS Unit 3 Cycle 6 emergency core cooling system (ECCS) performance analysis, as required by item 1, Section 3, of the staff's SE of ABB-CE's topical report, CEN-372-P. Although the bounding 14 X 14 steamline break is not applicable, DNB propagation was examined for postulated accidents and was verified not to occur, as described in more detail below.

Section 6.9.1.10 of the TS lists the analytical methods previously reviewed and approved by NRC to determine the core operating limits. Plant operation is limited in accordance with the values of cycle-specific parameter limits that are established using these NRC-approved analytical methods. This TS amendment will be implemented in order to support startup of PVNGS Unit 3 following the refueling outage scheduled to be completed on November 27, 1995. For PVNGS Units 1 and 2, this amendment will be implemented before startup from their next refueling outages (Refueling 6, scheduled to begin September 21, 1996, and March 16, 1996, for Units 1 and 2, respectively).

2.0 Proposed Changes to TS Section 6.9.1.10

The licensee proposes to modify the list of analytical methods used to determine core operating limits, identified in TS Section 6.9.1.10, by adding the following references:

6.9.1.10.j ABB-CE "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

6.9.1.10.k Letter from A.C. Thadani (NRC) to A. E. Scherer (ABB-CE), dated April 10, 1990, ("Acceptance for Reference ABB-CE Topical Report CEN-372-P").

3.0 EVALUATION

The proposed amendment to TS Section 6.9.1.10 would add a new methodology to the analytical methods used to determine the core operating limits. The NRC staff has approved the use of the analytical method CEN-372-P-A (as documented in a letter dated April 10, 1990, from Ashok C. Thadani, Director, Division of Systems Technology, Office of Nuclear Reactor Regulation, to A.E. Scherer, Director, Nuclear Licensing, Combustion Engineering, "Safety Evaluation of Combustion Engineering Topical Report CEN-372-P-A, (Fuel Rod Maximum Allowable Gas Pressure,")) as an acceptable basis for the new fuel rod internal pressure criterion. The NRC approval letter directs licensees to (1) prepare a plant-specific LOCA analysis to determine the impact of maximum calculated rod pressure on cladding rupture timing and peak cladding temperatures and (2) analyze DNB propagation in postulated accidents if the bounding 14 x 14 steamline break is not applicable for calculating maximum cladding rupture strain and percent flow blockage.

ECCS Performance

ABB/CE evaluated the maximum calculated fuel rod pressure in the PVNGS Unit 3 Cycle 6 (U3C6) ECCS performance analysis. The peak cladding temperature of the ECCS performance analysis is determined by the large-break LOCA analysis. The staff has reviewed and approved CENPD-132, Supplement 3-P-A, June 1985, which sets forth the ABB-CE large-break evaluation model relied on to implement CEN-372-P-A. The large-break LOCA analysis for U3C6 was performed with the NRC-approved model. This is the same version of the model that was used to perform the large-break LOCA reference cycle analysis.

In order to evaluate the impact of the maximum calculated fuel rod gas pressure on ECCS performance for Cycle 6, STRIKIN-II cases were run at burnups ranging from 1,000 MWd/MTU, the burnup corresponding to the maximum initial fuel stored energy, to 61,000 MWd/MTU, the highest burnup analyzed in the fuel performance analysis. The specific burnups analyzed were selected on the basis of an evaluation of the fuel's stored energies and gas pressures calculated by FATES3B. The STRIKIN-II analysis explicitly considers the impact of high pressure fuel rod gas on the timing of cladding rupture and, consequently, on the cladding temperature at and above the location of cladding rupture.

The following four cases summarize results for four burnups analyzed for Cycle 6. The four burnups presented are:

Case 1	1,000 MWd/MTU	Burnup with the maximum initial fuel stored energy
Case 2	26,300 MWd/MTU	Highest burnup that can sustain the peak linear heat generation rate
Case 3	40,500 MWd/MTU	Burnup that combines a gas pressure near the critical gas pressure and a maximum linear heat generation rate near the peak linear heat generation rate
Case 4	61,000 MWd/MTU	Highest burnup analyzed in the fuel performance analysis

The results show that the high initial fuel gas pressure for Cases 2 and 3 cause cladding to rupture earlier than for Case 1, the maximum stored-energy case. Despite having the highest initial fuel gas pressure, Case 4 had a cladding rupture time comparable to that of Case 1 because of the significantly reduced peak linear heat generation rate at 61,000 MWd/MTU. The analysis shows that Case 1 results in the highest peak cladding temperature. By comparing the results for the limiting burnup for Cycle 6, Case 1, with the results for the reference cycle analysis, APS's analysis shows that the results for Cycle 6 are bounded by those of the reference cycle.

DNB Propagation

The bounding cladding strain calculated for the 14 x 14 steamline break was applied to the PVNGS 16 x 16 fuel. The potential for DNB propagation was mechanistically evaluated by means of the ABB-CE INTEG computer code which calculates the fuel cladding strain as a function of time. The fuel cladding strain model for the INTEG computer program is described in CEN-372-P-A.

The following assumptions were made in the INTEG models:

(1) Cladding temperature instantaneously reached the value predicted by the Condie-Bengston correlation at the onset of DNB.

(2) To maximize the amount of ballooning, the circumferential cladding temperature variation was neglected in applying the strain rate model.

(3) The internal gas pressure of the fuel rod is unaffected by cladding ballooning.

A parametric study established the time necessary to reach the cladding strain limit by varying differential rod pressure, local heat flux, local quality, and local mass flux. The ranges were selected to range from a normal value to a more conservative one.

The following ranges were used:

Heat flux	250 to 800E+3 Btu/hr-ft^2
Mass flux	1.4 to 2.5E+6 lbm/hr-ft^2
Quality	-0.1 to 0.4
Differential Pressure	700 to 1200 psid

The parametric study cases were run until the total circumferential strain reached the limit of 29.3 percent or 1,000 seconds, whichever came first. The results were organized as tables of time to reach the strain limit and, to facilitate interpolation, graphical dependence of time to the strain limit for each of the four independent variables. Therefore, the time to reach the cladding strain limit may be determined for any combination of thermal hydraulic parameters.

For any postulated accident, the most limiting value for each of the thermal hydraulic parameters is determined from the transient results at the axial location of DNB. This set of conditions is used to enter the parametric figures and a time to the strain limit is determined.

The time necessary to reach the strain limit is compared with the time limit that the fuel rod is calculated to be in DNB. If the time in DNB for a given transient is less than the time to reach the cladding strain limit, DNB will not propagate.

For a sheared shaft event, as an example, the time to reach the strain limit using bounding values for each of the four thermal-hydraulic parameters was 60 seconds. Since the fuel is in DNB for only 5 seconds, DNB will not propagate. Comparisons of postulated accidents for PVNGS U3C6 yield the same conclusion. Future PVNGS cycles will be similarly compared and analysis performed as required. Using the described methodology, DNB propagation in ABB-CE 16 x 16 fuel does not occur for PVNGS.

In conclusion, the impact of the maximum pressure of fuel rod gas calculated for PVNGS Unit 3, Cycle 6 was evaluated as part of the PVNGS Unit 3, Cycle 6 ECCS performance analysis. Except for the highest burnup analyzed, the time of cladding rupture decreased as the initial pressure of fuel rod gas increased with burnup. However, the peak cladding temperature occurred at the burnup with the maximum initial fuel stored energy. The analysis also revealed that the ECCS performance for PVNGS Unit 3, Cycle 6 is bounded by that of the reference cycle analysis. The evaluation also demonstrated that the degree of cladding deformation is no more than the limit defined by the topical report on fuel rod maximum pressure (CEN-372-P-A). Thus, DNB is shown not to propagate.

The staff and the licensee discussed several of the assumptions stated in the July 31, 1986, safety evaluation of CE's model for ECCS large-break response. The model is applicable to CE designs supplied with CE-manufactured Zircaloy fuel. All of the PVNGS units utilize CE-manufactured Zircaloy fuel. PVNGS Unit 3 has a TS allowance to substitute up to a total of 80 fuel rods clad with zirconium-based alloys for in-reactor performance through Fuel Cycle 6. These rods are in bundles in non-limiting positions in the core and are, therefore, acceptable for use with this new methodology (ABB-CE topical report CEN-372-P-A). The staff also verified that the licensee adheared to the following assumptions in the staff's original safety evaluation:

(1) When the heat transfer coefficients resulting from the HCROSS computer program are greater than those resulting from the FLECHT-based correlation, the FLECHT values are utilized.

(2) The limiting break flow discharge coefficient has been determined by an appropriate break spectrum (three guillotine and three slot breaks).

(3) Although the homogenous equilibrium break flow model was discussed in the topical, the Appendix K Moody model was used for predicting break flows.

(4) An axial power shape similar to Shape B was utilized.

Because the amendment will add CEN-372-P-A and because the NRC's letter of April 10, 1990, approves it for the "Administrative Controls" portion of the TS, this change to TS Section 6.9.1.10 is considered to be administrative in nature. Plant operation will continue to be limited in accordance with values of cycle-specific parameters established using NRC-approved methodologies.

The staff has reviewed the proposed changes to TS Section 6.9.1.10 and, based on this evaluation, finds them acceptable for operation of Palo Verde Nuclear Generating Station Units 1, 2 and 3, respectively.

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 relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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