Docket No. 50-528

November 19, 1993

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 AMENDMENT FOR PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1 (TAC NO. M87839)

The Commission has issued the enclosed Amendment No. 72 to the Facility Operating License No. NPF-41 for Palo Verde Nuclear Generating Station, Unit No. 1. The amendment consists of a change to the Technical Specifications (TS) in response to your application dated September 8, 1993.

The amendment adds a methodology supplement entitled, "System 80™ Inlet Flow Distribution," to the list of methods used to determine the core operating limits. The associated staff Safety Evaluation allows the application of a thermal margin penalty to the addressable constants for Unit 1, cycle 5.

A copy of the related Safety Evaluation is also enclosed. A notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original Signed by

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

Enclosures:
1. Amendment No. 72 to NPF-41
2. Safety Evaluation

cc w/enclosures:
See next page

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Arizona Public Service Company

Palo Verde

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
License No. NPF-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by 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 September 8, 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 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:

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(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 and must be fully implemented no later than 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 19, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 72 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 vertical lines indicating the areas of change. The corresponding overleaf page is also provided to maintain document completeness.

REMOVE	INSERT
6-20a	6-20a

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. "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 1, February 1975 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculational Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P, Supplement 2-P, July 1975 (Methodology for Specification 3.2.1, Linear Heat Rate).

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

i. "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).

j. 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: 6.9.1.10e, 6.9.1.10f, 6.9.1.10h.

k. Letter: O. D. Parr (NRC) to A. E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model Changes). NRC approval for: 6.9.1.10g.

l. 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.i.

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

SYSTEM 80 INLET FLOW DISTRIBUTION - SUPPLEMENT 1-P

TO ENCLOSURE 1P TO LD-82-054

RELATED TO AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-41

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. STN 50-528

1.0 INTRODUCTION

By letter September 8, 1993, Arizona Public Service Company (APS or the licensee) submitted a request for a change to the Palo Verde Nuclear Generating Station, Unit No. 1 Technical Specifications (TS). The Arizona Public Service Company submitted this request on behalf of itself and 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 change would add a methodology supplement entitled, "System 80™ Inlet Flow Distribution," to the list of methods used to determine the core operating limits.

2.0 BACKGROUND

Enclosure 1P to LD-82-054, "Statistical Combination of Uncertainties - of System Parameter Uncertainties in Thermal Margin Analyses for System 80," is used to calculate the departure from nucleate boiling ratio (DNBR) for the Palo Verde Nuclear Generating Station (Palo Verde) units. This method has been previously reviewed and approved by the NRC. In a letter of March 30, 1993 (Ref. 1), Arizona Public Service Company (APS) requested U.S. Nuclear Regulatory Commission (NRC) review of Supplement 1-P to Enclosure 1-P to LD-82-054, "System 80™ Inlet Flow Distribution." This revised supplement to the methodology for assessing core thermal margin treats core inlet flow distribution data in a statistical manner.

3.0 EVALUATION

Supplement 1-P to Enclosure 1P to LD-82-054 describes a revised System 80 core inlet flow distribution for use with ABB-CE Statistical Combination of Uncertainties (SCU) methodology for assessing core thermal margin. The revised core inlet flow distribution and the associated uncertainties are the

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result of re-evaluating the System 80 reactor flow model data. The re-evaluation treats the core inlet flow distribution data in a statistical manner, as opposed to the deterministic method currently used. The objective of the revised methodology is to reduce conservatism in the current deterministic approach to gain additional calculated core thermal margin.

The report summarizes the current deterministic method of treating the core inlet flow data as an introduction to the revised methodology. The geometric features of the core lower support structure are described in relation to their possible impacts on core inlet flow distribution. Based on these features, the core inlet plane is regionalized to account for potential differences in inlet flow rates. The System 80 flow model data is then used to determine the core inlet flow factors and their uncertainties for the core inlet regions. A statistical test is applied to the resulting core inlet flow factors to support the hypothesis that the selected regions have separate, distinct core inlet flow factors.

APS currently uses a deterministic method to account for the inlet flow distribution and associated uncertainties in thermal margin analysis. The revised approach applied to determine the System 80 core inlet flow distribution is to identify regional flow factors based on the assumption that the inlet flow for groups of assemblies is determined by some common relationship to the upstream flow geometry. It consists of the following steps:

- Define geometric features in the reactor vessel which influence the core inlet flow distribution.
- Examine the core inlet flow distribution data to identify regions which have similar inlet flow factors, and categorize those regions.
- Determine the mean inlet flow factor and standard deviation of the flow factors for each region.
- Test the null hypothesis that the mean inlet flow factor values for two regions are from the same population. If that hypothesis can be rejected at a significance level of 5% for an equal-tails test, assume that the mean flow factors for the two regions are distinct values.
- Repeat the test pairwise for each of the regions that have been identified.

The use of this approach resulted in six different regions, each with its own mean value and standard deviation for inlet flow factors (as identified in Table 3-2 of Reference 1). The licensee's report concludes that these revised flow factors, along with the sample standard deviations of the flow factors, are a valid set to be used for future System 80 thermal margin licensing analyses.

For the System 80 plants, the current SCU methodology treats the core inlet flow distribution in a deterministic manner. As a result of the reassessment of the System 80 core inlet flow distribution, the core inlet flow distribution and its associated uncertainties will be treated in a statistical manner.

The system parameter SCU methodology consists of developing a minimum departure from nucleate boiling ratio (MDNBR) response surface which provides the functional relationship between the dependent variable, MDNBR, and the independent system parameter variables. The response surface is used to combine the probability density functions (PDFs) or the uncertainties associated with each of the independent variables into a resultant DNB PDF. The DNB PDF is then used to determine the MDNBR value at the 95% probability and 95% confidence levels. The 95/95 SCU MDNBR limit is used in conjunction with a best estimate thermal margin model to assess margin to the DNBR limit.

With respect to the System 80 SCU methodology described in Reference 2, the revised core inlet flow distribution approach will involve the following changes:

- The sensitivity of DNBR with respect to inlet flow factor will be determined for the limiting assembly and adjacent assemblies. These sensitivities will then be used with the appropriate inlet flow factor uncertainties to calculate an overall root-sum-square system parameter uncertainty, using the method presented in Section 5.3 of Reference 3.

- This approach will yield an increased MDNBR limit which will include allowances for uncertainties in hot assembly and adjacent assembly inlet flow. The increase in MDNBR limit can be accommodated directly by increasing the limit, or by applying a thermal margin penalty.

Reference 1 has proposed a set of factors to construct a best estimate core thermal margin model. Uncertainties in inlet flow to the hot assembly and adjacent assemblies can be accounted for statistically by either increasing DNBR or applying a thermal margin penalty using approved SCU methods. Uncertainties in inlet flow have been treated statistically using these methods (Ref. 3), and have been approved previously by the staff for other Combustion Engineering plants.

The licensee has proposed to implement the uncertainties associated with using a best estimate inlet flow by applying a thermal margin penalty to the addressable constants for Palo Verde Unit 1, cycle 5, as described in References 4 and 5. This has been reviewed by the staff and is acceptable for use in Palo Verde Unit 1 during cycle 5.

Based on the above, the staff has concluded that the proposed change to add a methodology supplement entitled, "System 80™ Inlet Flow Distribution," to the list of methods used to determine the core operating limits (as described in the TS change request dated September 8, 1993,) is acceptable. Additionally, in a meeting with the staff on October 6, 1993, the licensee has committed, to submit a generic implementation of flow uncertainties utilizing an

increased MDNBR for Units 2 and 3, and latter cycles of Unit 1. The NRC staff has allowed the use of applying a thermal margin penalty to the addressable constants for Unit 1, cycle 5. However, the implementation of this methodology is served best by increasing the MDNBR setpoint. The licensee plans to submit a generic application addressing a revised MDNBR setpoint for all three units. The NRC must review this generic implementation before it is used.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 53583). The amendment also changes recordkeeping or reporting requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (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 amendment.

6.0 CONCLUSION

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

Principal Contributor: L. Tran
B. Holian
G. Schwenk

Date: November 19, 1993

REFERENCES

1. Letter from W. F. Conway (APS) to Document Control Desk (NRC), "Supplement 1-P to Enclosure 1-P to LD-82-054," March 30, 1993.

2. NUREG-0852, Supplement 2, CESSAR System 80 SSER 2, dated September 1983.

3. "Statistical Combination of Uncertainties, Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One Unit 2," CEN-139(P), November 1980.

4. "Impact of Improved Inlet Flow Distribution on Typical CPC Overall Uncertainty Factor," CEN-421(v)-P, July 1993.

5. "Presentation to NRC on Impact of Improved Inlet Flow Distribution on the CPCS," CEN-424(v)-P, October 1993.