February 1, 2000

Mr. Gregg Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
P. O. Box 53999
Phoenix, AZ 85072-3999

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENTS RE: CHANGES TO LOW PRESSURE SAFETY INJECTION TRAIN ALLOWED OUTAGE TIMES (TAC NOS. MA4915, MA4916 AND MA4917)

Dear Mr. Overbeck:

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications in partial response to your application dated February 26, 1999, as supplemented by letter dated May 21, 1999.

The amendments extend the allowed outage time for one low pressure safety injection train from 72 hours to 7 days. The staff was unable to process the request contained in your application to delete Appendix D of the operating licenses since this issue was not addressed in the initial *Federal Register* notice dated April 7, 1999. The staff plans to include your request to delete Appendix D in our review of your December 1, 1999, letter requesting updates to the operating licenses.

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,

/***RA/***

Mel Fields, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 124 to NPF-41
2. Amendment No. 124 to NPF-51
3. Amendment No. 124 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

February 1, 2000

Mr. Gregg Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
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Enclosures:
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4. Safety Evaluation

cc w/encls: See next page *Please note document needs to be scan into Adams. There are tech spec. pages.

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ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
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 February 26, 1999, as supplemented by letter dated May 21, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 124, 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 shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/***RA/***

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 1, 2000

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
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 February 26, 1999, as supplemented by letter dated May 21, 1999, 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. 124, 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 shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/***RA/***

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: February 1, 2000

<u>ARIZONA PUBLIC SERVICE COMPANY, ET AL.</u>

<u>DOCKET NO. STN 50-530</u>

<u>PALO VERDE NUCLEAR GENERATING STATION, UNIT 3</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 124
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 February 26, 1999, as supplemented by letter dated May 21, 1999, 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. 124, 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 shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/***RA/***

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment numbers and contains a marginal line indicating the area of change.

REMOVE	INSERT
3.5.3-1	3.5.3-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 124

TO FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By application dated February 26, 1999, as supplemented by letter dated May 21, 1999, Arizona Public Service Company (APS or the licensee) requested changes to the Technical Specifications (TSs) for the Palo Verde Nuclear Generating Station (Palo Verde), Units 1, 2, and 3. APS 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 amendments would allow extensions of the allowed outage time for one low-pressure safety injection (LPSI) train from 72 hours to 7 days. This will allow greater flexibility in the scheduling and implementation of maintenance on the subject equipment, and avoid potential unscheduled plant shutdowns or requests for temporary relief for non-risk-significant conditions.

The May 21, 1999, supplement provided clarifying information that was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Since the mid-1980s, the NRC has been reviewing and granting improvements to TSs that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it...

> expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA (probabilistic safety assessment)[1] or risk survey and any available literature on risk insights and PSAs....Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical

[1]PSA and PRA are used interchangeably herein.

> Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decisionmaking and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

In May 1995, the Combustion Engineering Owners Group (CEOG) submitted several Joint Application Reports for the staff's review. One of the CEOG Joint Application Reports provided justifications for extensions of the TS completion time for the LPSI system.[2] The justifications for this extension are based on a balance of probabilistic considerations, traditional engineering considerations, including defense-in-depth, and operating experience. Risk assessments for all of the Combustion Engineering (CE) plants are contained in the reports. The staff first reviewed the Joint Application Reports and then reviewed the licensee's plant-specific amendment request, which incorporated the Joint Application Reports by reference.

Arkansas Nuclear One, Unit 2 (ANO-2) had been the lead CE plant for the LPSI system TS changes. The staff performed an in-depth review of the ANO-2 PRA methodology relating to these changes, as the lead plant for all of the CEOG. Therefore, a portion of the review of the Palo Verde amendment request was based on a comparison of the Palo Verde PRA results with those from ANO-2.

3.0 EVALUATION

The staff evaluated the licensee's proposed amendments to extend the TS completion time (completion time and allowed outage time [AOT] are used interchangeably herein) for one LPSI train out of service from 72 hours to 7 days using insights derived from traditional engineering

[2] CE NPSD-995, "Joint Application Report for Low Pressure Safety Injection System AOT Extension," May 1995.

considerations and the use of PRA methods to determine the safety impact of extending the completion times.

Traditional Engineering Evaluation

The current Palo Verde TSs address the LPSI system as a portion of the emergency core cooling system (ECCS). The two trains of the LPSI system, in combination with the two trains of the high-pressure safety injection (HPSI) system, form two redundant ECCS trains. TS 3.5.3 requires two ECCS trains to be operable. If one or more ECCS trains are inoperable, and at least 100% of the ECCS flow equivalent to a single operable ECCS train is available, the inoperable components must be returned to operable status within 72 hours or a plant shutdown is required.

The proposed change will allow up to 7 days for the licensee to restore operability to an inoperable LPSI train that is the cause of ECCS train inoperability. In some instances, corrective maintenance of the LPSI pump and valves and testing of valves may require taking one train of LPSI out of service for more than several days. Thus, repair within the existing completion time cannot be ensured and may result in an unscheduled shutdown or a request for temporary relief to allow continued plant operation. On the basis of the review of maintenance requirements of the LPSI train for CE pressurized water reactors (PWRs), the licensee determined that a 7-day completion time would provide sufficient margin to effect most anticipated preventive and corrective maintenance activities and LPSI train valve surveillance tests at power.

The primary role of LPSI trains during power operation is to contribute to the mitigation of a large loss-of-coolant accident (LOCA). The frequency of a large LOCA event is on the order of 10^{-4} per year. In contrast, during Modes 5 and 6, the operability of at least one LPSI train is required at all times for reactor coolant system (RCS) heat removal. Thus, in the broad view, performing preventative and corrective maintenance at power on LPSI trains can contribute to an overall enhancement of plant safety by increasing the availability of the LPSI train for shutdown cooling (SDC) during Modes 5 and 6, when it is most needed.

Another role of the LPSI system is defining the end state for a design-basis steam generator tube rupture (SGTR) event. In this design-basis event, the HPSI functions to keep the core covered at all times, and the LPSI system is required to effect SDC and thereby terminate the event. SDC is initiated after the break has been isolated and the radioactive releases have been controlled.

In the event that one LPSI train is out of service and the second LPSI train fails, the operator can continue to control the SGTR event by steaming of the unaffected steam generator. Even though loss of both LPSI trains is beyond the design-basis accident assumptions, this cooling mechanism can be maintained indefinitely, provided condensate is available to the unaffected steam generator. Without considering condensate storage tank refill, Palo Verde has sufficient inventory to steam the affected steam generator for more than 24 hours. Therefore, having one LPSI train out of service should not affect the licensee’s ability to mitigate an SGTR event.

Probabilistic Risk Assessment Evaluation

The staff used a three-tiered approach to evaluate the risk associated with the proposed TS changes. The first tier evaluated the PRA model and the impact of the completion time extensions for the LPSI system on plant operational risk. The evaluation of the PRA model relied, in part, on a cross comparison approach with a similar plant. The second tier addressed the need to preclude potentially high risk configurations, by identifying the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration during the time when one LPSI train is out of service.

The third tier evaluated the licensee's configuration risk management program to ensure that the applicable plant configuration will be appropriately assessed from a risk perspective before entering into or during the proposed completion times. Each tier and the associated findings are discussed below.

Tier 1 Evaluation

After completing a detailed evaluation for the tentative approval of LPSI TS AOT extension for Arkansas Nuclear One, Unit 2 (ANO-2), the original CEOG lead plant for the risk-informed TS pilot project, the staff used a cross comparison approach to consider the viability of similar AOT relaxations for other participating CEOG plants, including Palo Verde. The pilot technical evaluation report[3] used in support of the staff's draft safety evaluation for ANO-2[4] focused on:

- the process adopted by the CEOG to assess single AOT risk,
- the identification of ANO-2 accident sequences in which credit was taken for safety injection tanks and LPSI,
- independent verification of the single AOT risk [essentially equivalent to incremental conditional core damage probability (ICCDP)[5]], and
- determination of the significance of single AOT risk relative to an acceptance guideline value.

The objective of this cross comparison evaluation is to use insights derived from the ANO-2 technical evaluation to examine the validity of the conclusions drawn in the joint submittals. The staff believes that the findings of the lead pilot plant evaluation will be generally applicable to other CE plants, due to the fact that a common methodology was employed by the CEOG to

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[3]SCIE-NRC-318-97, "Technical Evaluation of Combustion Engineering Owners Group (CEOG) Joint Application for Safety Injection Tanks and Low Pressure Safety Injection System Allowed Outage Time (AOT) Extension," July 21, 1997.

[4]SECY-97-095, "Probabilistic Risk Assessment Implementation Plan Pilot Application for Risk-Informed Technical Specifications," April 30, 1997.

[5]ICCDP = [(conditional CDF with the subject equipment out of service) - (baseline CDF with nominal expected equipment unavailabilities)] X (duration of single AOT under consideration).

quantify AOT risk and CE plants have similar design characteristics. The staff confirmed that differences in the underlying PRA models are chiefly attributed to:

- minor design differences,
- operational differences,
- success criteria assumptions, and
- common cause failure β-factor or multiple Greek letter (MGL) assumptions.

The cross comparison draws on information contained in the CEOG Joint Application Reports, the licensees' responses to the staff's requests for additional information, the licensees' individual plant examinations (IPEs) performed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and the corresponding IPE evaluations performed by the staff.

The following factors are chiefly responsible for the differences in LPSI AOT risks among the CE plants:

- use of LPSI to mitigate multiple initiating events,
- HPSI redundancies, and
- LPSI common cause β-factor or MGL assumptions

Based on the licensee's February 26, 1999, submittal, the staff estimates that the LPSI preventive and corrective maintenance weighted average single AOT risk for Palo Verde is 9.3E-08 and is less than the acceptance guideline value 5.OE-07 from Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." Further, the staff believes that this estimate is reasonable since the conjoint frequency of large break LOCA, safety injection tank malfunction, and deleterious break location is extremely small. In addition, the change in the Palo Verde updated baseline core damage frequency (CDF) (as reported in the licensee submittal of 2/26/99) due to the LPSI AOT change is about 0.5%,i.e., from 3.93-E-05 per year to 3.95E-05 per year. The change in CDF of 2.OE-07 per year is within the acceptance guidelines published in RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The staff concludes that the approach and findings obtained for ANO-2, and the cross comparisons to other CE plants, are generally applicable to Palo Verde. To complete the first tier evaluation, the staff reviewed the quality of the Palo Verde PRA and how the LPSI system was modeled.

The staff conducted an audit of the licensee's PRA model on June 23, 1999, and the information discussed below was acquired by the staff during this audit. The licensee stated that the original IPE PRA was done using the CAFTA/SETS software. The 1994 update was done with Crystal and then with IRRAS software. The 1998 update involved a complete review of the PRA (now in Risk Spectrum software), concentrating on Human Reliability Analysis, Common Cause Failures, and Initiating Event Frequencies; this review involved consultants. The 1994 update added gas turbine generators, operator recovery actions for non-essential auxiliary feedwater, Bayesian-updated initiating event frequencies and data (resulting in a large reduction in turbine trip contributions), and changes in common cause failure rates of HPSI pumps. This update (1994) resulted in a baseline CDF of 4.7E-05/yr, compared to that for the IPE of 1.0E-04/yr.

The current PRA model includes removal of the dependence of the turbine-driven auxiliary feedwater pump on room ventilation, inclusion of gas turbine generator power, removal of loss of control room ventilation as an initiating event, changes from test-interval to demand failure rates for many important components (large pumps, valves), improved Bayesian-updating methodology, addition of Transition Risk methodology, and Bayesian-updated unavailability, failure rate, and initiating event frequency values.

The licensee states that, in the area of Quality Assurance, its PRA meets the RG 1.174 requirements, with qualified personnel, control of documentation/independent review, records, audit function, and corrective action. The licensee has an industry-accredited training program for engineering support personnel, which includes initial training and job-specific training.

The licensee has established a system for controlling PRA procedures. The PRA model and analyses are documented as engineering studies and, as such, are subjected to the existing engineering process, including revision control and independent review. The computer software is controlled by procedure also. All engineering studies are maintained as a lifetime record, as is software installation history.

The original IPE was subjected to independent consultant review and to CEOG cross-comparison review (with 12 other CE plants). In 1998, there was an internal independent review, as well as special external reviews by ERIN Engineering for Level I (Human Reliability Analysis, Common Cause Failures, Initiating Event Frequencies) and Level II (Complete Model). A CEOG PRA certification was completed in September 1999.

In the area of corrective action , the normal plant corrective action process is used. PRA model errors are screened, including software errors and analysis errors.

During the site audit, NRC staff members discussed the PRA modeling of the LPSI system operation with licensee personnel. The licensee stated that the PRA models credit the LPSI in responding to large LOCAs, small LOCAs, and SGTR events. Palo Verde is a General Design Criterion 4 “leak before break” plant, which means that an analysis reviewed and approved by the Commission demonstrated that the probability of a large primary system piping rupture is extremely low. Following a small LOCA with successful HPSI injection but unsuccessful switchover to HPSI recirculation, the LPSI system in recirculation alignment was credited as an alternative to high-pressure recirculation. Medium LOCAs require hot leg injection to prevent boron precipitation if the break is in the cold leg piping. It is not possible to align LPSI to inject into the hot leg so LPSI is not credited as an alternative to high-pressure recirculation for medium LOCAs. The licensee stated that steam generator safety and atmospheric dump valves do not normally open following an SGTR. In sequences where the steam generator safety or dump valves are challenged, open successfully, but fail to close, alignment of the LPSI system in SDC mode is credited to provide long-term cooling to the depressurized core. The system could also be credited in the same operating mode for other LOCAs outside containment but these sequences are not modeled in the PRA. Thus, the LPSI system is used only to mitigate rare initiating events (aside from its normal SDC function). It is used only as a primary mitigating system for the very unlikely large LOCA initiating event. For the other low frequency initiators, the system is used only as a backup following equipment and human action failures.

The staff finds that the small ICCDP estimated for the change in AOT from 3 to 7 days is consistent with the credit taken for the system in the PRA modeling, and that the extensive licensee review of the PRA models during the 1992-1998 reconstitution provides reasonable assurance that the models appropriately reflect the equipment and procedural characteristics at the plant.

This completes the staff's first tier evaluation of the licensee's proposal to extend the completion time for one LPSI train from 3 to 7 days. Based on the above discussion, the staff finds acceptable the PRA model used by the Palo Verde licensee and also concludes that there is minimal impact of the completion time extensions for the LPSI system on plant operational risk.

Tier 2 Evaluation

The licensee did not identify any dominant risk-significant configurations associated with the proposed LPSI train completion time extension. The licensee states that a review of large early release scenarios for the CE PWRs indicates that early releases arise as a result of the following class of scenarios:

1. Containment Bypass Events.

   These events include interfacing system LOCAs and SGTRs with a concomitant loss of steam generator isolation (e.g., stuck open main steam safety valve).

2. Severe Accidents Accompanied by Loss of Containment Isolation.

   These events include any severe accident in conjunction with an initially unisolated containment.

3. Containment Failure Associated with Energetic Events in the Containment.

   Events causing containment failure include those associated with the High-Pressure Melt Ejection (HPME) phenomena (including direct containment heating (DCH)) and hydrogen conflagrations/detonations.

Of the three radioactive release categories associated with the above event categories, Class 1 tends to represent a large early release of potentially direct, unscrubbed fission products to the environment. Class 2 events encompass a range of releases varying from early to late that may or may not be scrubbed. Class 3 events result in a high-pressure failure of the containment, typically immediately upon or slightly after reactor vessel failure. Detailed Level 2 analysis for the plant condition with one LPSI train inoperable was not performed. However, assessment of the expected change in the large early release fraction was made by assessing the impact of the availability of the LPSI system on the above event categories.

1. Containment Bypass Events

Events contained in this category that may rely on the LPSI for event mitigation include the large interfacing-systems LOCA (ISLOCA) (i.e., failure of an SDC line). Testing and/or maintenance of containment isolation valves residing in the LPSI system are governed under the plant TSs. Thus, no change in the ISLOCA frequency is expected.

ISLOCAs are characterized by continuous and unreplenished loss of RCS inventory and makeup. In these scenarios, core damage ultimately results following the depletion of reactor coolant. Thus, provided that a continuous independent water supply is not available during the accident, the ISLOCA will progress into early core damage regardless of LPSI availability.

2. Severe Accidents Accompanied by Loss of Containment Isolation

Another event contributing to large early fission product releases could occur when an unmitigated large LOCA occurs in conjunction with an initially unisolated containment. Significant fission product releases would not occur unless the containment atmosphere is unscrubbed (that is, sprays are inoperable). This latter combination of events is considered of very low probability and would not significantly increase with a decrease in LPSI pump availability, because LPSI is not a major support system for the containment sprays.

3. Containment Failure Associated with Energetic Events in the Containment

Class 3 events are dominated by RCS transients that occur at high pressure. These events exclude those where LPSI system performance would be called for and therefore, LPSI status is not a contributor to this event category. It is, therefore, concluded that increased unavailability of the LPSI system (as could potentially result as a consequence of an increased AOT) will have a very small impact on the large early release fraction for CE PWRs.

External events can potentially lead to high risk configurations and therefore are included in the second tier evaluation. The LPSI at-power function is largely to mitigate large LOCA events. The external events of fire, severe weather, and flooding are not considered to be initiators of large LOCA events. The only external events that need to be considered are seismic events.

In letter number 102-03407, dated June 30, 1995, APS submitted to NRC the Individual Plant Examination of External Events (IPEEE) for Palo Verde. Section 3 of the IPEEE documents an extensive evaluation of seismic events. Subsection 3.1.3.2, “RLE Response Spectra Conclusions,” concluded the following concerning the Review Level Earthquake (RLE):

> Based upon the comparison of the RLE spectra and the design basis In-Floor Response Spectra (IFRS), it can be concluded that all seismic category 1 equipment on the IPEEE Safe Shutdown Equipment List, including relays and equipment required for containment performance, was qualified to a higher level of seismic demand than required by IPEEE. The seismic qualification process assures that seismic category 1 equipment, including relays, will perform their intended function during and after the RLE (as well as for the design basis earthquake). This confidence level is greater than would be achieved by meeting only the industry screening criteria.

Thus, the staff concurs that large LOCA is not considered to be a credible consequence of a seismic event for Palo Verde, and that extending the completion time for a LPSI train will not increase plant risk where external phenomena are the initiating events.

The Tier 2 evaluation did not identify the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration.

Tier 3 Evaluation

The licensee proposes to implement a Configuration Risk Management Program (CRMP) and to establish the CRMP requirements in the Technical Requirements Manual,which is included in the Updated Final Safety Analysis Report by reference and controlled by 10 CFR 50.59. The purpose of the CRMP is to ensure that a proceduralized PRA-informed process is in place that assesses the overall impact of plant maintenance on plant risk.

Implementation of the CRMP will enable appropriate actions to be taken or decisions to be made to minimize and control risk when performing on-line maintenance for systems, structures, and components (SSCs) with a risk-informed completion time.

The scope of the SSCs included in the CRMP are those SSCs modeled in the licensee's plant PRA in addition to those SSCs considered of high safety significance per RG 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2 (the Maintenance Rule regulatory guide), that are not modeled in the PRA.

The content of the CRMP process consists of the following components:

a. Provisions for the control and implementation of a Level 1 at-power internal events PRA-informed methodology. The assessment is to be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the plant configuration described by the Limiting Conditions for Operation (LCO) Action Statement for preplanned activities.

c. Provisions for performing an assessment after entering the plant configuration described by the LCO Action Statement for unplanned entry into the LCO Action Statement.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment-out-of service conditions while in the plant configuration described by the LCO Action Statement.

e. Provisions for considering other applicable risk-significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

Key Element 1. Implementation of CRMP

The intent of the CRMP is to implement Section (a)(3) of the Maintenance Rule (10 CFR 50.65) with respect to on-line maintenance for risk-informed TSs, with the following additions and clarifications:

a. The scope of the SSCs to be included in the CRMP will be those SSCs modeled in the licensee's plant PRA in addition to those SSCs considered of high safety significance per RG 1.160, Revision 2 (the Maintenance Rule regulatory guide), that are not modeled in the PRA.

b. The CRMP assessment tool is PRA informed, and may be in the form of either a risk matrix, an on-line assessment, or a direct PRA assessment.

c. The CRMP will be invoked as follows for:

   Risk-Informed Inoperability: A risk assessment will be performed prior to entering the LCO Condition for preplanned activities. For unplanned entry into the LCO Condition, a risk assessment will be performed in a time frame consistent with the plant's Corrective Action Program.

   Additional SSC Inoperability and/or Loss of Functionality: When in the risk-informed Completion Time, if an additional SSC within the scope of the CRMP becomes inoperable/non-functional, a risk assessment shall be performed in a time frame consistent with the plant's Corrective Action Program.

d. Tier 2 commitments apply for planned maintenance only, but will be evaluated as part of the Tier 3 assessment for unplanned occurrences.

Key Element 2. Control & Use of the CRMP Assessment Tool

a. Plant modifications and procedure changes will be monitored, assessed, and dispositioned.

   - Evaluation of changes in plant configuration or PRA model features can be dispositioned by implementing PRA model changes or by the qualitative assessment of the impact of the changes on the CRMP assessment tool. This qualitative assessment recognizes that changes to the PRA take time to implement and that changes can be effectively compensated for without compromising the ability to make sound engineering judgments.

   - Limitations of the CRMP assessment tool are identified and understood for each specific Completion Time extension.

b. Procedures exist for the control and application of CRMP assessment tools, including description of the process when outside the scope of the CRMP assessment tool.

Key Element 3. Level 1 Risk-Informed Assessment

The CRMP assessment tool is based on a Level 1, at power, internal events PRA model. The CRMP assessment may use any combination of quantitative and qualitative input. Quantitative assessments can include reference to a risk matrix, pre-existing calculations, or new PRA analyses.

a. Quantitative assessments should be performed whenever necessary for sound decisionmaking.

b. When quantitative assessments are not necessary for sound decisionmaking, qualitative assessments will be performed. Qualitative assessments will consider applicable, existing insights from quantitative assessments previously performed.

Key Element 4. Level 2 Issues/External Events

External events and Level 2 issues are treated qualitatively and/or quantitatively.

Guidance for implementing the CRMP is provided by plant procedures.

The licensee also has the ability to analyze the risk impact of outage configurations in a timely manner using a tool called the Plant Configuration Risk Indicator Matrix.

The staff's third tier evaluation concludes that the risk-informed CRMP proposed by the licensee will satisfactorily assess the risk associated with the removal of equipment from service during the proposed LPSI AOT. The program provides the necessary assurances that appropriate assessments of plant risk configurations, including during outage conditions, are sufficient to support the completion time extension request for the LPSI system.

Summary

The staff has evaluated the licensee's proposed changes for compliance with regulatory requirements as documented in this evaluation and has determined that they are acceptable. This determination is based on the following:

1. The traditional engineering evaluation reveals that increasing the availability of the LPSI system for SDC during outages by performing preventive and corrective maintenance at power can contribute to an overall enhancement of plant safety.

2. The staff finds acceptable the PRA model used by the Palo Verde licensee and also concludes that there is minimal impact of the completion time extensions for the LPSI system on plant operational risk (Tier 1 evaluation) .

3. The review of potentially high risk configurations did not identify the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration (Tier 2 evaluation).

4. The risk-informed CRMP proposed by the licensee will satisfactorily assess the risk associated with the removal of equipment from service during the proposed LPSI AOT (Tier 3 evaluation).

5. The licensee has stated that the maintenance rule (10 CFR 50.65) will be the vehicle that controls the actual equipment maintenance cycle by defining unavailability performance criteria for the LPSI system. The AOT extension will allow efficient scheduling of maintenance within the boundaries established by implementing the maintenance rule. The maintenance rule will thereby be the vehicle that monitors the effectiveness of the AOT extension. Application of these implementation and monitoring strategies will help to ensure that extension of the TS AOT for the LPSI system does not degrade operational safety over time and that the risk incurred when a LPSI system is taken out of service is minimized.

The staff, therefore, finds that the completion time for one LPSI train may be extended to 7 days, with a negligible impact on risk.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 17023). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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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