UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

November 9, 2005

James M. Levine, Executive Vice
President, Generation
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000528/2005004, 05000529/2005004; AND 05000530/2005004

Dear Mr. Levine:

On September 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palo Verde Nuclear Generating Station, Units 1, 2, and 3, facility. The enclosed integrated report documents the inspection findings, which were discussed on October 4, 2005, with you and other members of your staff.

The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The report documents six NRC identified findings and two self-revealing findings. Six of these findings were evaluated under the risk significance determination process as having very low safety significance (Green). One finding was not suitable for evaluation under the significance determination process; however, it was determined to be of very low safety significance (Green) by NRC management review. One finding impeded the regulatory process and was assessed in accordance with the NRC Enforcement Policy. Seven findings involved violations of NRC requirements. Because of the very low safety significance of these violations and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. Five licensee identified violations, which were determined to be of very low safety significance, are listed in Section 4OA7 of this report. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Troy W. Pruett, Chief
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Division of Reactor Projects

Dockets: 50-528
50-529
50-530

Licenses: NPF-41
NPF-51
NPF-74

Enclosure:
NRC Inspection Report 05000528/2005004, 05000529/2005004, and 05000530/2005004
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SISP Review Completed: _TWP__ ADAMS: / Yes ☐ No Initials: TWP_____
/ Publicly Available ☐ Non-Publicly Available ☐ Sensitive / Non-Sensitive

R:_REACTORS_PV\2005\PV2005-04RP-GGW.wpd

RIV:RI:DRP/D	RI:DRP/D	SRI:DRP/D	SPE:DRP/D
JFMelfi	PLBenvenuto	GGWarnick	GEWerner
T-TWP	**T-TWP**	**T-TWP**	**NA**
11/08/05	11/08/05	11/08/05	10/ /2005

C:DRS/PSB	C:DRS/OB	C:DRS/EMB	C:DRS/PEB	C:DRP/D
MPShannon	AGody	CPaulk	LJSmith	TWPruett
/RA/	**/RA/**	**/RA/**	**/RA/**	**/RA/**
11/14/05	11/3/05	11/02/05	11/04/05	11/09/05

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-528, 50-529, 50-530
Licenses:	NPF-41, NPF-51, NPF-74
Report:	05000528/2005004, 05000529/2005004, 05000530/2005004
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Tonopah, Arizona
Dates:	July 1 through September 30, 2005
Inspectors:	G. Warnick, Senior Resident Inspector J. Melfi, Resident Inspector P. Benvenuto, Resident Inspector J. Clark, PE, Chief, Engineering Branch 1 P. Elkmann, Emergency Preparedness Inspector C. Paulk, Senior Reactor Inspector N. Taylor, Project Engineer, Project Branch D S. Alferink, Reactor Inspector
Approved By:	Troy W. Pruett, Chief, Branch D Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000528/2005004, 05000529/2005004; 05000530/2005004; 07/01/05 - 09/30/05; Palo Verde Nuclear Generating Station, Units 1, 2 and 3; Integrated Resident and Regional Report; Nonroutine Evolutions, Operability Evaluations, Post Maintenance Testing, Identification and Resolution of Problems.

This report covered a 3-month period of inspections by three resident inspectors, two reactor inspectors, one emergency preparedness inspector, one project engineer, and one engineering branch chief. The inspection identified seven noncited violations and one finding. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management's review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a was identified as a result of the licensee’s failure to monitor leakage properly using the spent fuel pool leak detection surveillance as required by Procedure 40DP-9OPA3, “Area 3 Operator Logs, Modes 1-4.” This resulted in leakage of spent fuel pool water through two adjacent concrete walls. Specifically, operations personnel did not monitor the spent fuel pool telltale drains for evidence of leakage for a period of five and a half months, and failed to take the necessary action to reschedule the task. This issue involved human performance crosscutting aspects associated with operations personnel following procedures and having a questioning attitude. This issue also involved problem identification and resolution crosscutting aspects associated with operations and engineering personnel implementing timely corrective actions. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2814209.

  The finding is greater than minor because it affects the equipment performance and human performance attributes of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This finding cannot be evaluated by the significance determination process because Manual Chapter 0609; “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and Appendix G; “Shutdown Operations Significance Determination Process,” do not apply to the spent fuel pool . This finding is determined to be of very low safety significance by NRC management review because radiation shielding was provided by the spent fuel pool water level, the spent fuel pool cooling and fuel building ventilation systems were available, and there were multiple sources of makeup water. Additionally, there were no adverse effects to the environment because the small amount of leakage (8 ounces) could not reach the local perched or regional ground water due to their distances (70 and 300 feet, respectively) (Section 1R14).

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a was identified as a result of the licensee's failure to follow Procedure 40OP-9ZZ04, "Plant Startup Mode 2 to Mode 1," and Procedure 40OP-9FT01, "Feedwater Pump Turbine A," which resulted in an automatic reactor trip and main steam isolation signal due to a high steam generator water level. Specifically, the secondary reactor operator failed to: (1) ensure downcomer feed flow to both steam generators, (2) properly setup the controller, and (3) establish a stable steam generator level between 30 to 40 percent prior to placing the feedwater controller in automatic. Additionally, the secondary reactor operator failed to inform the control room supervisor and other control room personnel when he made numerous transfers into and out of automatic valve control to make manual feedwater adjustments when attempting to recover steam generator water level. This issue involved human performance crosscutting aspects associated with operations personnel following procedures and attention to detail. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2825485.

  The finding is greater than minor because it affects the human performance attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. A Phase 2 analysis was required because the Phase 1 Worksheet in Manual Chapter 0609, "Significance Determination Process," determined that the finding affected the initiating events cornerstone and contributed to the likelihood that mitigation equipment or functions would not be available. Using the Phase 2 worksheets associated with transients and transients without the power conversion system, the finding is determined to have very low safety significance since all remaining mitigation capability was available or recoverable (Section 1R14).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding involving poor work controls due to ineffective and inaccurate technical communications between organizations. During maintenance on the Unit 1 high pressure safety injection long-term recirculation check Valve SIAV522 operations, maintenance, and engineering personnel did not implement management expectations specified in Procedure 93DP-OLC07, "10 CFR 50.59 and 72.48 Screenings and Evaluations," The Nuclear Engineering Strategic Plan, and Procedure 40 DP-90P26, "Operability Determination." Specifically, (1) the licensee did not verify the accuracy of an engineer's statement regarding 10 CFR 50.59 documents, and consequently, did not ensure that all documents used to support the work activity existed prior to the commencement of work. (2) Maintenance personnel changed the freeze seal location without consulting operations or engineering, even though the location was a key assumption that formed the basis for several conclusions in the operability evaluation. This change required a revision to the opperability evaluation before work could start. (3) Engineering personnel incorrectly informed operations personnel that only 5 to 10 gpm was needed to full stroke Valve SIAV522 when approximately 100 gpm was needed. This issue required a change to the instructions provided to operations prior to testing the check valve. The issue involved human performance crosscutting aspects associated with inadequate communications between the engineering, maintenance, and operations organizations. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Requests 2822343 and 2831411.

The finding is greater than minor since it could become a more significant safety concern in that the failure to provide accurate information to support operational decision making could result in improper operability determinations. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because it only affected the mitigating systems cornerstone and did not result in the loss of safety function of a single train or system for greater than the Technical Specification allowed outage time (Section 1R15).

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a for the failure of maintenance personnel to follow Procedure 31DP-9ZZ01, “Lubricant Sampling,” and Work Order 2724849. Specifically, a maintenance technician incorrectly determined that the oil sample taken from the Unit 2 high pressure safety injection pump was satisfactory, when the oil sample did not meet the acceptance criteria. Consequently, immediate actions to address potential equipment deficiencies were not taken until the samples were analyzed by a lubrication engineer approximately two weeks later. This finding involved human performance crosscutting aspects associated with maintenance personnel following procedures and attention to details. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2828545.

  The finding is greater than minor since the failure to follow the lubricant sampling procedure, if left uncorrected, would become a more significant safety concern in that degraded equipment conditions may not be identified and corrected in a timely manner. A Phase 2 analysis was required because the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet determined that there was a loss of the long term cooling safety function of a single train of high pressure safety injection for greater than the Technical Specification allowed outage time. A senior reactor analyst determined that the high pressure safety injection pump was only required to operate for 24 hours to meet the assumptions necessary in the risk model to preclude sequences that result in core damage. Consequently, this finding is determined to have very low safety significance (Section 1R19).

- Green. The inspectors identified two examples of a noncited violation of Technical Specification 5.4.1.a for the failure to follow Procedure 77ST-9SB19, “CPCS Channel Functional Test,” and Work Order 2824743 during core protection calculator software installation. Specifically, maintenance technicians: (1) failed to change the software loading instructions of Work Order 2824743 prior to proceeding with the core protection calculator software installation when it could not be used as written, and (2) failed to follow the surveillance test procedure used to perform a core protection calculator functional test. This finding involved human performance crosscutting aspects associated with instrumentation and controls personnel following procedures. This finding also involved problem identification and resolution crosscutting aspects associated with instrumentation and controls personnel identifying degraded or nonconforming conditions. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2825189.

The finding is greater than minor since it could become a more significant safety concern in that the failure to follow procedures when performing maintenance and testing on safety related equipment could result in an unintentional actuation or impact the ability of the equipment to perform its required function. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because it only affected the mitigating systems cornerstone and did not result in the loss of safety function of a single train or system for greater than the Technical Specification allowed outage time (Section 1R19).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to correct a discrepancy between the current condition of the boronometer and the required configuration described in the Updated Final Safety Analysis Report. Specifically, in April 2003 the licensee identified the need to perform a Licensing Document Change Request and a corresponding 10 CFR 50.59 screening due to the abandonment of the Updated Final Safety Analysis Report required boronometer, but failed to implement corrective actions to ensure that the Licensing Document Change Request and 10 CFR 50.59 screening were performed. This issue involved problem identification and resolution crosscutting aspects associated with engineering personnel implementing timely corrective actions. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2823704.

  The finding is greater than minor because it was associated with the design control performance attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the reliability and availability of systems that respond to initiating events. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because there was no actual loss of safety function (Section 4OA2).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the improper control of design parameters for the ex-core nuclear instrument safety channels in that engineering personnel did not correctly translate design requirements, nor did they properly control design basis information regarding ex-core safety channels. Additionally, Technical Specification required values were maintained apart from design calculations and documents. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2612092.

  This finding is greater than minor because if left uncorrected it could become a more significant safety concern in that failures to maintain design calculations could result in the incorrect setting of safety related devices. The finding is associated with the mitigating systems cornerstone. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because there was not an actual loss of safety function. (Section 4OA2).

- SLIV. The inspectors identified a noncited Severity Level IV violation of 10 CFR 50.9 for providing incomplete or inaccurate information to the NRC. Specifically, the licensee provided incomplete and inaccurate information regarding the design control of ex-core safety channel log power instrument setpoints. This information was determined to be material in that it affected the NRC's ability to determine compliance with NRC requirements. This issue was entered into the licensee's corrective action program as Condition Report/Disposition Request 2829051.

  This finding was not assessed via NRC Manual Chapter 0609, "Significance Determination Process," because the licensee's actions impeded the regulatory process. Therefore, this finding was assessed in accordance with the NRC Enforcement Policy. The finding is associated with the mitigating systems cornerstone. The inspectors determined that engineering personnel had additional information, including the subsequently corrected revision of the calculation going through final verification, and additional explanatory setpoint procedures, which were not referenced or provided during the original correspondence by the licensee. Had the complete and accurate information been supplied at the time of the original request in 2003, the NRC would have identified a design control violation at that time. The safety consequence of this issue is of very low safety significance, in that there was no actual loss of a safety function (Section 4OA2).

B. Licensee-Identified Violations

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full power until August 12, 2005, when the unit was shutdown as required by Technical Specification (TS) 3.8.1 to troubleshoot and rework the emergency diesel generator Train B voltage regulator. Several equipment issues delayed unit startup including a reactor coolant pump (RCP) oil seal, a pressurizer auxiliary spray valve, and the control element assembly drive system. On August 26, while performing a plant startup, a reactor trip and main steam isolation occurred due to a high steam generator water level. The unit returned to essentially full power on August 31 and remained there for the duration of the inspection period.

Unit 2 operated at essentially full power until August 22, 2005, when the unit was shutdown as required by TS 3.0.3 due to an issue identified with the core protection calculator (CPC) software. Following resolution of the CPC software issue, the unit returned to essentially full power on August 29 and remained there for the duration of the inspection period.

Unit 3 operated at essentially full power until July 5, 2005, when the unit was shutdown due to an equipment issue with an RCP oil seal. Following repairs to the oil seal, the unit returned to essentially full power on July 14, and remained there for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

Readiness for Impending Adverse Weather Conditions

The inspectors completed a review of the licensee's readiness for impending adverse weather involving monsoon conditions. The inspectors: (1) evaluated implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions; (2) verified that operator actions defined in the licensee’s adverse weather procedure maintains readiness of essential systems and that adequate operator staffing was specified; (3) reviewed maintenance records to determine that applicable surveillance requirements were current before the anticipated monsoon developed; and (4) reviewed plant modifications, procedure revisions, and operator work arounds to determine if recent facility changes challenged plant operation.

- August 4, 2005, Units 1, 2, and 3, implementation of Procedure 40AO-9ZZ21, “Acts of Nature,” Revision 21, due to monsoon conditions

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns

The inspectors: (1) walked down portions of the two below listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned and (2) compared deficiencies identified during the walkdown to the licensee's corrective action program (CAP) to ensure problems were being identified and corrected.

- September 7, 2005, Unit 2, high pressure safety injection (HPSI) system Train A while Train B was out of service for maintenance

- September 21, 2005, Unit 1, emergency diesel generator Train A while Train B was out of service for maintenance

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Routine Inspection

The inspectors walked down the six below listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures are commensurate with the significance of the deficiency; and (7) reviewed the CAP to determine if the licensee identified and corrected fire protection problems.

- July 19, 2005, Unit 1, spent fuel pool (SFP) building
- August 3, 2005, Unit 3, auxiliary building, all accessible elevations
- August 31, 2005, Unit 3, fuel building, all accessible elevations
- September 7, 2005, Unit 2, fuel building, 100, 120, and 140 foot elevations
- September 8, 2005, Unit 3, main steam support structure, all accessible elevations
- September 21, 2005, Unit 3, control room building, all accessible elevations

The inspectors completed six samples.

Annual Inspection

The inspectors observed a fire brigade drill on July 12, 2005, to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) the number of personnel assigned to the fire brigade; (2) use of protective clothing; (3) use of breathing apparatuses; (4) use of fire procedures and declarations of emergency action levels; (5) command of the fire brigade; (6) implementation of pre-fire strategies and briefs; (7) access routes to the fire and the timeliness of the fire brigade response; (8) establishment of communications; (9) effectiveness of radio communications; (10) placement and use of fire hoses; (11) entry into the fire area; (12) use of fire fighting equipment; (13) searches for fire victims and fire propagation; (14) smoke removal; (15) use of pre-fire plans; (16) adherence to the drill scenario; (17) performance of the post-drill critique; and (18) restoration from the fire drill. The licensee simulated a fire in the 140 foot elevation of the Unit 1 turbine building.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

Semi-annual Internal Flooding

The inspectors: (1) reviewed the Updated Final Safety Analysis Report (UFSAR), the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; (2) reviewed the CAP to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding could reasonably achieve the desired outcomes; and (5) walked down the below listed areas to verify the adequacy of (a) equipment seals located below the floodline, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

C August 23 - 30, 2005, Units 1, 2, and 3, emergency core cooling system pump rooms

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a steam generator tube rupture with high reactor coolant system (RCS) activity.

- September 14, 2005, Scenario SES-0-04-I-02, “Dropped CEA/High RCS Activity/SGTR,” Revision 0

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the two below listed maintenance activities to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the Maintenance Rule, Appendix B of 10 CFR Part 50, and TSs.

- September 23, 2005, Unit 1, completed evaluation of water in emergency diesel generator Train A governor that caused a failure to start on March 17, 2005, as documented in CRDR 2782680

- September 30, 2005, Units 2 and 3, completed review of target rock solenoid valve failures associated with the auxiliary feedwater and main steam systems as documented on CRDRs 2775099, 2799930, 2817153, and 2825162

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

Risk Assessment and Management of Risk

The inspectors reviewed the below listed assessment activities to verify:
(1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- July 5, 2005, Units 1 and 3, evaluation of the risk management action levels during performance of Procedure 73ST-9AF02, “AFA-P01 - Inservice Test,” Revision 33; “AFB-P01-Inservice Test,” Revision 15; and Procedure 73ST-9XI38, “AFW Pumps Discharge Check Valves - Inservice Test,” Revision 12

- August 31, 2005, Unit 3, auxiliary feedwater pump Train A removed from service to replace the governor power supply per Work Mechanism 2809442

The inspectors completed two samples.

Emergent Work Control

The inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the CAP to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- July 8, 2005, Unit 1, HPSI Train A hot leg injection check Valve SIAV522 leakage described in CRDR 2813941

- July 7, 2005, Unit 3, replacement of RCP 1A seal package due to excessive leakage as described in Work Order (WO) 2813212

- July 29, 2005, Unit 1, troubleshooting of ultrasonic flow meter following receipt of an intermittent alarm documented in CRDR 2826472

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Evolutions and Events (71111.14, 71153)

a. Inspection Scope

The inspectors: (1) reviewed operator logs, plant computer data, and/or strip charts for the below listed evolutions to evaluate operator performance in coping with nonroutine events and transients; (2) verified that the operator response was in accordance with the response required by plant procedures and training; and (3) verified that the licensee has identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the nonroutine evolutions sampled.

- July 2, 2005, Unit 1, fuse failure led to safety injection signal Leg 1-3 actuation. This actuation was from the logic portion of the Engineering Safety Features Actuation System, and caused the valve group relays to actuate. Only the valves received an actuation signal, and no pumps, fans, or diesel generators started. A loss of letdown occurred due to the closing of letdown inboard containment isolation Valve CHA-UV-516. The licensee entered Procedure 40AO-9ZZ17, "Inadvertent ESFAS," Revision 10, and Procedure 40AO-9ZZ05, "Loss of Letdown," Revision 14, for this actuation. This event was documented in CRDR 2812981.

- July 8, 2005, Unit 1, an auxiliary operator observed water seeping through the Unit 1 SFP south wall. Additional observations discovered water seeping through the outside of the east wall on the fuel building. The SFP telltale drains were opened, and approximately 1200 gallons of water were released from the drains. Upon further investigation it was established that the SFP liner telltale drains had not been checked for several months. This event was documented in CRDR 2814209.

- August 26, 2005, Unit 1, while performing a plant startup, a reactor trip and main steam isolation occurred due to a high steam generator water level. The high steam generator water level resulted from a mismatch between feed and steam flow following feedwater transfer from the non-essential feedwater pump to the main feedwater pump. This event was documented in CRDR 2825485.

The inspectors completed three samples.

b. Findings

.1 Failure to Monitor Telltale Drains Resulted in Spent Fuel Pool Leakage

Introduction. A Green self-revealing noncited violation (NCV) of TS 5.4.1.a was identified as a result of the licensee's failure to properly implement the SFP leak detection surveillance. This resulted in leakage of SFP water through two adjacent concrete walls.

Description. On July 8, 2005, during a routine area tour, an auxiliary operator in the Unit 1 fuel building observed water seeping from the SFP south wall at the 105 foot elevation in the cleanup pump area. There were also white deposits that looked like solidified boric acid. Upon further observation a second leak was discovered outside of the fuel building at the 104 foot elevation of the SFP east wall. Licensee personnel obtained samples of the water and debris outside of the Unit 1 fuel building and identified trace quantities of radioactive Cobalt-60, Antimony-125, and Cesium-137. Samples of the leakage indicated that the source of the water was from the SFP since boron concentrations were consistent with SFP chemistry. Following the discovery of the leakage, the SFP telltale drains were opened and approximately 1200 gallons of water were released from the drains. The licensee removed all contamination from the outer SFP walls that resulted from the leakage and ensured that no residual activity remained outside of the fuel building. The licensee reviewed the potential environmental impacts from the condition and determined that no adverse effects resulted from the SFP leakage because the small amount of leakage outside the SFP building (8 ounces) could not reach the local perched or regional groundwater due to their distances (70 and 300 feet, respectively) below the ground surface. Additionally, the structural impacts to the concrete were minimal because the leakage only occurred once and the drying of the borated water in the concrete wall would stop continued corrosion of the reinforcing bar.

Procedure 40DP-9OPA3, “Area 3 Operator Logs, Modes 1-4,” Revision 52, required that auxiliary operators open the telltale drains daily to monitor for SFP liner leakage. The logs specified that if water flow is observed for more than two minutes, the system engineer must be contacted to investigate. Additionally, the logs state that if a weekly reading cannot be done, it must be rescheduled with the control room supervisor (CRS) or shift manager. Further investigation identified that the SFP liner telltale drains had not been checked for a period of five and a half months. The auxiliary operators stopped opening the drains on January 31, 2005, due to blockage in the drain line that directs flow from the basin that collects the telltale drain discharge to the fuel building sump. The auxiliary operators informed the control room of the condition and Work Mechanism 2773782 was initiated to repair the drain. Subsequently, the work mechanism was inadvertently closed prior to corrective maintenance and prior to resumption of the required daily telltale drain readings. The licensee attributed the work control errors to a lack of understanding of the significance of the issue and inadequate prioritization by management.

Analysis. The performance deficiency associated with the finding was the failure to follow the procedure to monitor the SFP telltale drains for evidence of leakage. The finding is greater than minor because it affects the equipment performance and human performance attributes of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This finding cannot be evaluated by the significance determination process because Manual Chapter 0609; “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and Appendix G; “Shutdown Operations Significance Determination Process,” do not apply to the SFP. This finding is determined to be of very low safety significance by NRC management review because radiation shielding was provided by the SFP water level, the SFP cooling and fuel building ventilation systems were available, and there were multiple sources of makeup water. This issue involved human performance crosscutting aspects associated with operations personnel following procedures and questioning attitude. This issue also involved problem identification and resolution crosscutting aspects associated with operations and engineering personnel implementing timely corrective actions.

Enforcement. Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 8.a, requires procedures for leak detection systems tests. Procedure 40DP-9OPA3, “Area 3 Operator Logs, Modes 1-4,” Revision 52, Appendix B, Page 18, required in part, “(1) if a weekly reading or preventive maintenance cannot be done when scheduled, then reschedule with the CRS or SM, and document performance date in comments section, and (2) if water flow is observe from a telltale drain for more than two minutes, then contact the system engineer to investigate.” Contrary to this, operations personnel did not monitor the telltale drains for evidence of leakage for a period of five and a half months, and failed to take the necessary action to reschedule the task. Because the finding is of very low safety significance and has been entered into the CAP as CRDR 2814209, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2005004-01, “Failure to Monitor Telltale Drains Resulted in Spent Fuel Pool Leakage to the Environment.”

.2 Reactor Trip and Main Steam Isolation

Introduction. A Green self-revealing NCV of TS 5.4.1.a was identified for the licensee’s failure to follow procedures which resulted in an automatic reactor trip and main steam isolation signal due to a high steam generator (SG) water level.

Description. On August 26, 2005, Unit 1 had completed a reactor startup and was in Mode 2 at approximately 3 percent power. The licensee was conducting a plant startup following a short notice outage (Section 1R20) per Procedure 40OP-9ZZ04, “Plant Startup Mode 2 to Mode 1,” Revision 48. The secondary reactor operator performed the SG feedwater transfer from non-essential feedwater Pump AFN-P01 to main feedwater pump Train A using Procedure 40OP-9FT01, “Feedwater Pump Turbine A,” Revision 14, and placed the feedwater regulating valve controllers in automatic. The secondary reactor operator noticed that SG-1 and SG-2 levels had lowered from approximately 30 percent to approximately 13 and 16 percent, respectively, a short time after the transfer. The secondary reactor operator took manual control of the feedwater control system in an attempt to recover SG levels. The secondary reactor operator’s actions resulted in a mismatch between feedwater flow and steam flow which established an excessive feedwater flow rate. The secondary reactor operator was not able to stabilize SG water level before SG-1 reached the Hi SG-1 Level Setpoint of 91.5 percent narrow range level, which initiated a reactor trip and main steam isolation signal. The control room staff performed standard post trip actions and diagnosed a reactor trip. Post trip heat removal was accomplished by the operation of atmospheric dump valves and auxiliary feedwater Pump B since the secondary (e.g., power conversion system) was not available due to the main steam isolation signal. The event was subsequently classified as an uncomplicated reactor trip with no emergency classification required. The unit was stabilized in Mode 3.

The post trip investigation conducted by the licensee determined that the secondary reactor operator did not establish the conditions required by Procedure 40OP-9FT01 when placing the feedwater valves into automatic control. Specifically, the secondary reactor operator failed to: (1) ensure downcomer feed flow to both SGs, (2) properly set up the controller, and (3) establish a stable SG level between 30 to 40 percent. Consequently, both downcomer feedwater valves were in the closed position with both SG levels decreasing when the controller was placed in automatic. Upon discovery of the lowering level, the secondary reactor operator further complicated the level transient by making a series of manual feedwater adjustments and transfers into and out of automatic valve control, which resulted in an unrecoverable over feed condition. The licensee's investigation also determined that the secondary reactor operator took manual control of the controller and the subsequent actions to recover SG level without the knowledge of the CRS, contrary to the requirements of Procedure 40DP-9OP02, "Conduct of Shift Operations," Revision 31.

Analysis. The performance deficiency associated with this finding was the failure to follow procedures to place the main feedwater pump in service and to manually override an automatic control system. The finding is greater than minor because it affects the human performance attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the Phase 1 Worksheet in Manual Chapter 0609, "Significance Determination Process," the finding is determined to contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available, requiring Phase 2 analysis. The initiating event likelihood is determined to be less than 3 days since the finding is related to a brief evolution performed during a plant startup. Further, the initiating event likelihood was increased by one order of magnitude since the finding did not involve a support system, but rather, involved a human performance error when placing a feedwater pump in service. Using the worksheets associated with transients and transients without the power conversion system, the finding was determined to have very low safety significance since all remaining mitigation capability was available and/or recoverable. This issue involved human performance crosscutting aspects associated with operations personnel following procedures and operator attention to detail.

Enforcement. Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 3.k, requires procedures for operating the SG feedwater system. Procedure 40OP-9FT01, "Feedwater Pump Turbine A," Revision 14, Section 4.3, required that equipment be controlled and plant conditions established to ensure a smooth transition when placing the first main feedwater pump in service. Also, Procedure 40DP-9OP02, "Conduct of Shift Operations," Revision 31, Section 12, required that, concurrence shall be obtained from the CRS prior to placing any automatic controller in manual or returning a controller to automatic, and all control room personnel shall be informed of the status of the controller and changes in their expected actions in the event of a plant transient with the automatic controller in manual. Contrary to this, the secondary reactor operator failed to establish plant conditions for placing a feedwater pump in service and did not inform the CRS of controller manipulations. Specifically, the secondary reactor operator did not: (1) ensure downcomer feed flow to both SGs, (2) properly set up the controller, and (3) establish a stable SG level between 30 to 40 percent prior to placing the feedwater controller in automatic. Additionally, the secondary reactor operator failed to inform the CRS and other control room personnel when he made numerous transfers into and out of automatic valve control to make manual feedwater adjustments when attempting to recover SG water level. Because the finding is of very low safety significance and has been entered into the CAP as CRDR 2825485, this violation is being treated as an NCV,

consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000528/2005004-02, "Improper Control of Steam Generator Feedwater System Resulted in a Reactor Trip and Main Steam Isolation."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the UFSAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the significance determination process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- July 12, 2005, Unit 3, through wall leakage from reactor drain tank as noted in deficiency WO 2813864

- July 28, 2005, Units 1, 2, and 3, inadequate atmospheric dump valve nitrogen accumulator drop test and associated operability impact documented in CRDRs 2818612 and 2818836

- August 8, 2005, Unit 2, spray pond pump Train A indicated flow below required values as described in Operability Determination (OD) 300

- August 13, 2005, Unit 1, OD documented in CRDR 2822343 to justify freeze seal installation to support maintenance on HPSI long-term recirculation check Valve SIAV522

- September 11, 2005, Unit 2, OD 296, "Auxiliary Feedwater Pump AFA-P01 Governor DC Control Power Dropping Resister R-17," Revision 1

The inspectors completed five samples.

b. Findings

Introduction. A Green finding was identified by the inspectors for poor work controls due to ineffective and inaccurate technical communications between organizations.

Description. On August 13, 2005, the licensee developed an operability evaluation through CRDR 2822343 to determine if the application of a temporary alteration (freeze seal), in support of HPSI long-term recirculation check Valve SIAV522 maintenance, would impact the operability of the RCS loops and the shutdown cooling system. This

CRDR evaluation was performed to satisfy 10 CFR 50.65(a)(4) and ensure compliance with TS requirements. The evaluation concluded that the use of a freeze seal for the pressure boundary between the RCS and safety injection system did not impact the operability of the RCS loops and shutdown cooling system.

Procedure 93DP-0LC07, "10 CFR 50.59 and 72.48 Screening and Evaluations," Revision 8, Step 3.3.5, specified that the provisions of the NEI guidance documents that would allow making temporary alterations in support of maintenance without reviewing the proposed activity under the requirements of 10 CFR 50.59 are not applicable to Palo Verde. Rather, Step 3.3.5 contained a more restrictive administrative requirement that all temporary alterations be performed in accordance with approved procedures, including being reviewed in accordance with Procedure 93DP-0LC07. This requirement would be satisfied by using a specific procedure to control the temporary alteration installation that has been reviewed through the 10 CFR 50.59 process.

The inspectors reviewed CRDR 2822343, and the associated operability evaluation, and noted that the documents stated that application of the freeze seal and its impact on the class piping had been reviewed and approved by Piping Specification 13-PN-204, and that the change to the specification to allow freeze sealing was reviewed by 10 CFR 50.59. The inspectors requested a copy of the safety evaluation from the shift technical advisor just prior to commencement of the work activity. The inspectors were informed that the 10 CFR 50.59 review was not readily available since it had been performed years earlier and had been archived, and that a copy would be available for review on the next normal business work day. The licensee was not able to locate the referenced 10 CFR 50.59 review on the following business work day. The only document the licensee was able to locate was a 10 CFR 50.59 screening from 1988 associated with Revision 0 of the freeze seal procedure. The licensee evaluated the cause of the inaccurate information referenced in the operability evaluation through CRDR 2822343, and determined that the information was based on a statement made by engineering personnel. The engineer stated that he believed there was a change to Piping Specification 13-PN-204 to allow freeze sealing that was reviewed by 10 CFR 50.59, which was not the case. The licensee did not verify the accuracy of the engineer's statement, and consequently, failed to ensure that the documents used to support the work activity existed prior to the commencement of work.

The inspectors noted during review of the operability evaluation that the freeze seal would be installed on a vertical run of pipe, downstream of Valve SIAV522. The evaluation used the location as an underlying assumption throughout the analysis. The inspectors attended the pre-job briefing conducted prior to commencement of the maintenance activities. The maintenance supervisor described the work planned and indicated that the freeze seal had been installed on the first horizontal run of pipe, downstream of Valve SIAV522. The inspectors noted the location change and questioned the freeze seal supervisor why the location was changed from the location used in the operability evaluation. The freeze seal supervisor stated that he was not aware that an operability evaluation had been performed and that the change in location was necessary since a circumferential weld interfered with the original location. The freeze seal work group changed the location without consulting operations or engineering. The inspectors questioned operations and engineering personnel whether the operability evaluation was still valid since a key assumption that formed a basis for several conclusions had changed. The licensee stopped work and revised the operability evaluation to appropriately evaluate the operability impact using the correct location.

During the review of the retest requirements, the inspectors questioned the acceptability of testing methods proposed by the licensee following maintenance on Valve SIAV522. The inspectors reviewed Procedure 40OP-9SI02, “Recovery from Shutdown Cooling to Normal Operating Lineup,” Revision 60A, Section 17.0, that would be used to perform the forward flow test to full stroke open the valve as required by the in-service testing program. The inspectors noted that Step 17.3.11 directed operators to establish 80 to 120 gpm flow through the valve, with 100 gpm as the optimal value. The inspectors questioned the basis of the specified flow rates and whether the flow rate was adequate to full stroke open the valve. Operations personnel presented the inspectors with an e-mail from engineering personnel that stated, according to the Crane Handbook, only 5 to 10 gpm was needed to full stroke open the valve. Therefore, the planned forward flow test per Procedure 40OP-9SI02, Section 17.0, would satisfy in-service testing requirements. The inspectors performed an independent verification of the calculated flow rate necessary to full stroke open the valve using the Crane Handbook since the values specified by engineering personnel seemed inadequate. The inspectors determined that approximately 100 gpm was needed to full stroke open the valve, and challenged engineering regarding the values specified in the e-mail that was provided to operations. The licensee reviewed the Crane Handbook and concluded that the inspectors’ calculation was correct. Subsequently, engineering changed their instructions to recommended that operations flush Valve SIAV522 at a minimum flow rate of 100 gpm to satisfy retest requirements.

Analysis. The performance deficiency associated with this finding was poor work control processes due to a failure to communicate clear, factual, and accurate information between organizations. The finding is greater than minor since it could become a more significant safety concern in that the failure to provide accurate information to support operational decision making could result in improper ODs. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because it only affected the mitigating systems cornerstone and did not result in the loss of safety function of a single train or system. The issue involved human performance crosscutting aspects associated with inadequate communications between the engineering, maintenance, and operations organizations.

Enforcement. No violations of regulatory requirements occurred. The Nuclear Engineering 2005 Strategic Plan described the expectations for effective technical communications. The Strategic Plan stated, in part, that, “Effective technical communications within, and across, organizational lines is fundamental to our success. We will communicate across organizational boundaries to ensure appropriate understanding and actions are taken on technical issues. Both verbal and written communications will be clear and succinct as well as being timely, factual, and accurate." Contrary to this: (1) the licensee did not verify the accuracy of an engineer's statement regarding 10 CFR 50.59 documents, and consequently, failed to ensure that the documents used to support the work activity existed and satisfied procedural requirements, (2) maintenance personnel changed the freeze seal location without consulting operations or engineering, even though the location was a key assumption that formed the basis for several conclusions in the operability evaluation, and (3) engineering personnel incorrectly informed operations personnel that only 5 to 10 gpm was needed to full stroke Valve SIAV522 when approximately 100 gpm was needed. The inspectors determined that the finding did not represent a noncompliance because it involved the adequacy of communications between departments. This finding has been entered into the licensee's CAP as CRDRs 2822343 and 2831411, Finding (FIN) 05000528/2005004-03, "Communication Deficiencies Between Organizations."

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed the below listed operator workaround to: (1) determine if the functional capability of the system or human reliability in responding to an initiating event is affected; (2) evaluate the effect of the operator workaround on the operator's ability to implement abnormal or emergency operating procedures; and (3) verify that the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

- September 27, 2005, Unit 1, degraded power supply for annunciator lights on emergency diesel generator heating, ventilation, and air conditioning control

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the four below listed post maintenance test activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented. The inspectors also reviewed the CAP to determine if the licensee identified and corrected problems related to postmaintenance testing.

- August 4, 2005, Unit 2, retest following repair of HPSI pump Train B outboard bearing per WO 2820466

- August 14, 2005, Unit 1, retest following repair of emergency diesel generator Train B auto voltage regulator per WO 2821209

- August 15, 2005, Unit 1, retest following repair of HPSI check Valve SIAV522 per WO 2821956

- August 25, 2005, Unit 2, retest per Procedure 77ST-9SB19, “CPCS Channel C Functional Test,” Revision 3

The inspectors completed four samples.

b. Findings

.1 Untimely Assessment of Degraded HPSI Pump Oil Bearing Sample

Introduction. A Green NRC identified NCV of TS 5.4.1.a was identified by the inspectors for the failure of maintenance personnel to follow procedure requirements regarding examination of a lubricant sample.

Description. On July 20, 2005, maintenance personnel obtained an oil sample as part of a routine preventive maintenance activity on HPSI pump Train B per WO 2724849. The bearing oil sample was sent to the lubrication engineer for evaluation following completion of WO 2724849. On August 3, 2005, the lubrication engineer identified increased wear products in the bearing oil sample and concluded that there was not reasonable assurance that the HPSI pump would perform its design basis function of running continuously for 180 days. The CRS declared the Unit 2 HPSI pump Train B inoperable based on the results of the oil sample and recommendations from engineering. A subsequent engineering evaluation concluded that, although the pump was degraded for greater than the TS allowed outage time of 72 hours, the bearing would have lasted for at least two months of continuous pump operation.

The inspectors reviewed WO 2724849 and interviewed licensee personnel to determine the cause of the two week delay between the performance and evaluation of the bearing oil sample. Work Order 2724849 and Procedure 31DP-9ZZ01, “Lubricant Sampling,” Revision 6, required maintenance personnel to observe the sample for the following conditions: (1) wear metal, (2) visible dirt or other particles, (3) emulsification, or (4) a pungent or burnt odor. Section 3.4.1 of Procedure 31DP-9ZZ01 further stated that, “Initial sample examination is necessary to ensure immediate action is taken with samples that appear abnormal upon visual observation.” The inspectors reviewed color pictures of the oil sample and observed that the oil was significantly darker with observable particles in the oil. The mechanics incorrectly marked the oil sample as satisfactory on the completed WO 2724849, and the oil sample was not expedited for evaluation.

Analysis. The performance deficiency associated with this finding was that maintenance personnel did not implement the provisions of the lubrication sampling procedure or WO. The finding is greater than minor since the failure to follow the lubricant sampling process, if left uncorrected, would become a more significant safety concern in that degraded equipment conditions may not be identified in a timely manner. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to require a Phase 2 evaluation since HPSI pump Train A was degraded for greater than the TS allowed outage time. The Senior Reactor Analyst (SRA) determined that, although there is a design basis requirement of continuous pump operation for 180 days, the HPSI pump was only required to operate for 24 hours to meet the assumptions necessary in the risk model to preclude the sequences that result in core damage. Consequently, the finding is of very low safety significance. This finding involved human performance crosscutting aspects associated with maintenance personnel following procedures and attention to details.

Enforcement. Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 9a, requires maintenance that can affect safety-related equipment be properly preplanned and performed in accordance with written instructions appropriate to the circumstances. WO 2724849 and Paragraph 3.4.1 of Procedure 31DP-9ZZ01, "Lubricant Sampling," Revision 6, directed maintenance personnel to observe the sample for degraded conditions. Additionally, Procedure 31DP-9ZZ01 required immediate actions to be taken for a degraded oil sample. Contrary to this, on July 20, 2005, maintenance personnel performed an inadequate observation of an oil sample and did not implement immediate actions for a degraded oil sample. Consequently, immediate actions to declare HPSI Pump B inoperable and replace the bearings were not taken until the samples were analyzed approximately two weeks later. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2828545, this violation is being treated as an NCV consistent with Section V1.A of the NRC Enforcement Policy: NCV 05000529/2005004-04, "Improper Visual Analysis of Bearing Oil Sample."

.2 Failure to Follow Procedures During Core Protection Calculator Software Installation and Testing

Introduction. Two examples of a Green NCV of TS 5.4.1.a were identified by the inspector for the failure of instrumentation and control (I&C) personnel to adequately implement procedures for CPC software installation and retest. The examples involved: (1) a failure to change the WO prior to proceeding with CPC software installation when the WO could not be used as written, and (2) a failure to follow the surveillance test procedure used to perform a CPC functional test.

Description. On August 25, 2005, the inspectors observed the performance of WO 2824743 to install a CPC software update and the associated functional retest on Channel C. The unit was shutdown on August 22, 2005, to install the software update following notification by the CPC vendor that certain sensor failures or analog input module failures would not result in a corresponding trip of the CPC channel.

Work Order 2824743, Step 4.4.1.3, instructed the I&C technicians to reload the CPC Channel C Processor per the Common Q CPC System Software Installation Manual, Section 5. The inspectors observed that the software loading instructions contained several incorrect and missing steps, and that the task could not have been completed in accordance with the steps as written in the WO. However, the I&C technicians had worked through the WO inadequacies using skill of the craft techniques to complete the software installation. The inspectors asked the work group leader, who was also present to observe the maintenance activity, why the work was not stopped to correct the WO inadequacies when each deficiency was identified as required by the conduct of maintenance and procedure use and adherence procedures. The inspectors further questioned the leader why the software loading instructions had not been corrected prior to the software installation for CPC Channel C since the same WO had been used to load the software on CPC Channels A, B, and D the previous shift. The group leader acknowledged that the activity should have been stopped and the WO revised.

The inspectors reviewed the retest requirement specified in WO 2824743 which included performance of Procedure 77ST-9SB19, "CPCS Channel C Functional Test," Revision 3, Sections 7.0, 8.1, 8.5 through 8.18, and 9.0. The inspectors noted that these sections were the only portion of the surveillance test procedure included in the work package. The inspectors observed that I&C technicians would verify in Step 7.3 that Section 5.0, "Limitations and Precautions," and Section 6.0, "Personnel Indoctrination," had been read and understood by all test personnel. A copy of these additional sections were obtained and reviewed by the inspectors since they were not included in the work package. The inspectors noted that several items specified in Sections 5.0 and 6.0 were not completed. Additionally, the maintenance technicians commenced the retest using Procedure 77ST-9SB19, Section 7.0, and initialed that Step 7.3 had been completed without referencing Sections 5.0 or 6.0. Following questioning by the inspectors, the leader intervened and confirmed that the step had not been completed. The work group leader instructed the I&C technicians to perform Step 7.3 prior to continuing with the procedure.

Analysis. The performance deficiency associated with the examples of this finding was the failure of I&C technicians to follow procedures. The finding is greater than minor since it could become a more significant safety concern in that the failure to follow procedures when performing maintenance and testing on safety related equipment could result in an unintentional actuation or impact the ability of the equipment to perform required safety functions. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because it only affected the mitigating systems cornerstone and did not result in the loss of safety function of a single train or system. This finding involved human performance crosscutting aspects associated with I&C personnel following procedures. This finding also involved problem identification and resolution crosscutting aspects associated with I&C personnel identifying degraded or nonconforming conditions.

Enforcement. Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 9a, requires maintenance that can affect safety-related equipment be properly preplanned and performed in accordance with written instructions appropriate to the circumstances.

Procedure 30DP-9MP01, "Conduct of Maintenance," Revision 40, Step 3.3.4.2, required that, "Maintenance personnel will perform work instructions as per 01DP-0AP09, Procedure Use and Adherence." Procedure 01DP-0AP09, "Procedure Use and Adherence," Revision 0, Step 3.4.2, required, in part, "If the procedure cannot be used as written, change the procedure in accordance with an approved change method prior to proceeding with the activity."

The level of use designation for Procedure 77ST-9SB19, "CPCS Channel C Functional Test," Revision 3, is continuous. Procedure 01DP-0AP09 states that it is the responsibility of the procedure user to adhere to the requirements of the procedure in use. Procedure 01DP-0AP09 further states for continuous use procedures/sections, "When performing procedure steps: read and understand the step, self check and perform the step, check the action complete and expected response/results are received, and sign or check for completion of the step." Contrary to this, on August 25, 2005, I&C technicians failed to stop and change a procedure that could not be performed as written and failed to ensure procedure actions were completed before continuing with testing. Specifically, I&C technicians : (1) failed to change the software loading instructions of WO 2824743 prior to proceeding with CPC software installation when it could not be used as written, and (2) failed to follow the surveillance test procedure used to perform a CPC functional test. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2825189, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000529/2005004-05, "Failure to Follow Procedures During Core Protection Calculator Software Installation and Testing."

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant outage activities to verify defense in depth commensurate with the outage risk control plan and compliance with the TSs: (1) the risk control plan, (2) tagging/clearance activities, (3) RCS instrumentation, (4) electrical power, (5) decay heat removal, (6) SFP cooling, (6) inventory control, (7) reactivity control, (8) containment closure, (9) heatup and cooldown activities, and (10) licensee identification and implementation of appropriate corrective actions associated with outage activities.

- August 12 - 31, 2005, Unit 1, short notice outage to repair the auto voltage regulator on emergency diesel generator Train B. Several emergent equipment issues resulted in extending the short notice outage to repair HPSI hot leg injection check Valve SIAV522 internal back leakage, RCP 1A oil seal leak, and excessive pressurizer auxiliary spray valve leakage.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and TSs to ensure that the three below listed surveillance activities demonstrated that the SSC's tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- July 19, 2005, Unit 2, Procedure 36ST-9SE01, "Reactor Safety Channel Log Calibration," Revision 39

- August 5, 2005, Unit 2, inservice test per Procedure 73ST-9SG05, "ADV Nitrogen Accumulator Drop Test," Revision 21

- September 11, 2005, Unit 2, RCS leakage detection surveillance per Procedure 40ST-9RC02, "ERFDADS (preferred) Calculation of RCS Water Inventory," Revision 30

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the UFSAR, plant drawings, procedure requirements, and TSs to ensure that the below listed temporary modification was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability; (2) verified that the installation was consistent with the modification documents; (3) ensured that the post-installation test results were satisfactory and that the impact of the temporary modification on permanently installed SSC's were supported by the test; (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with temporary modifications.

- September 16, Unit 2, Temporary Modification 2390509, "Chemical Bin and Pump Skid and Tubing Routed to Each SP Pump Intake Structure"

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed in-office reviews of:

- Revision 6 to Emergency Plan Implementing Procedure 99, Appendices A and P, submitted June 29, 2005
- Revision 32 to the Palo Verde Nuclear Generating Station Emergency Plan, submitted July 12, 2005
- Revision 7 to EPIP 99, Appendix P, submitted August 11, 2005
- Revision 33 to the Palo Verde Nuclear Generating Station Emergency Plan, submitted August 26, 2005

These revisions:

- Revised EAL 1-1 to an initiating condition of 700°F as measured by core exit thermocouples, replacing 50°F superheat

- Revised EALs 8-4 and 8-5 to clarify that classifiable conditions exist as determined by the Emergency Coordinator, not the Emergency Operations Director

- Provided additional detail regarding the physical layout of the Operations Support Center(s)

The revisions were compared to their previous revisions, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, to NEI 99-01, “Methodology for Development of Emergency Action Levels,” Revision 2, and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the licensee adequately implemented 10 CFR 50.54(q).

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the below listed simulator-based training evolutions contributing to drill/exercise performance and emergency response organization performance indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” acceptance criteria.

- August 31, 2005, Unit 1 Simulator, Scenarios SES-0-09-AQ-00, “ECC Directed Turbine Unloading/Inadvertent MSIS/FRP (MVAC-3),” and SES-0-06-F-00, “Loss of PKB-M42/Loss of Vacuum/LOAF (restored by local operation of AFB-P01)”

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Reviews

In order to identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished by reviewing daily CRDR summary reports. The inspectors also reviewed daily summaries of work mechanisms initiated to determine whether CRDRs were generated as appropriate to properly evaluate potential maintenance rule impact, operability issues, and reportable conditions. No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspectors chose one issue for more in depth review to verify that licensee personnel had taken corrective actions commensurate with the significance of the issue. The issue and the basis for the selection are described below:

- CRDR 2596985, identification that the boronometer configuration was not in alignment with the UFSAR

When evaluating the effectiveness of the licensee's corrective actions for this issue, the following attributes were considered:

- Complete and accurate identification of the problem in a timely manner commensurate with its significance and ease of discovery
- Evaluation and disposition of operability and reportability issues
- Consideration of extent of condition, generic implications, common cause, and previous occurrences
- Classification and prioritization of the resolution of the problem commensurate with its safety significance
- Identification of root and contributing causes of the problem for significant conditions adverse to quality
- Identification of corrective actions which are appropriately focused to correct the problem
- Completion of corrective actions in a timely manner commensurate with the safety significance of the issue

b. Findings

Introduction. A Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified by the inspectors for the failure to correct a discrepancy between the current condition of the boronometer and required configuration described in the UFSAR.

Description. The boronometer is described in the UFSAR as a piece of equipment required to conform with the requirements of General Design Criteria 13 of 10 CFR Part 50, Appendix A. Specifically, Page 3-1-13 of the UFSAR stated the following:

> "A Boronometer, which determines the boron concentration in the reactor coolant by neutron absorption, is provided as a backup to the primary method of determining soluble poison concentration by routine sampling and analysis of reactor coolant."

There are many other references throughout the UFSAR to the boronometer in this regard. Reactor coolant system boron concentration is one piece of information required to complete Surveillance Requirement 3.1.3.1, which verifies that the overall core reactivity balance is within +/- 1.0 percent of predicted values while in Mode 1. Contrary to the description in the UFSAR, the licensee had effectively abandoned the boronometer in place. Preventive and corrective maintenance on the boronometer was historically completed using Procedure 36MT-9CH01, "Boronometer Calibration," which contained guidance for routine calibration and troubleshooting. Interviews conducted by the inspector with system engineers, I&C technicians, and operations personnel indicated that the boronometers had a history of poor reliability until approximately 2000, when maintenance ceased due to the unavailability of spare parts (specifically the high voltage power supply). The last successful performance of Procedure 36MT-9CH01 was: October 22, 1998, for Unit 1; January 12, 1998, for Unit 2; and November 20, 1998, for Unit 3. The abandonment of the boronometer left routine (daily) sampling as the only method available to determine RCS boron concentration.

On April 10, 2003, CRDR 2596985 was initiated to document that the boronometer configuration was not in conformance with the UFSAR. The response to the CRDR indicated that the boronometer would be removed by WO 2486234 beginning in refueling Outage 2R13. Condition Report Action Item (CRAI) 2608133 was initiated to perform a Licensing Document Change Request (LDCR) and a supporting 10 CFR 50.59 screening/evaluation to bring the station into compliance. The required LDCR and 10 CFR 50.59 screening were drafted on November 3, 2003, to support the planned WO. The CRDR and the associated CRAI were subsequently closed out to the WO on November 5, 2003. Condition Report/Disposition Request 2823704 was initiated on August 19, 2005, to report that the configuration discrepancy still existed and several new CRAIs were initiated to address the lack of conformance with the UFSAR and perform the required LDCR.

As of September 23, 2005, no LDCR or 10 CFR 50.59 screening had been performed to consider the effects of removing the boronometer from service. Paragraph 3.11.3 of Procedure 90DP-0IP10, "Condition Reporting," Revision 15 (which was in effect when CRAI 2608133 was closed out on November 5, 2003), stated:

> "A CRAI that requires a licensing document change to correct a deficiency shall only be closed to an approved LDCR..."

Contrary to this requirement, CRAI 2608133 was closed out to WO 2486234, which was never performed. As a result, the LDCR and 10 CFR 50.59 screening drafted for the WO were never implemented and the facility was not brought into compliance as the writers of the CRAI had intended.

Analysis. The deficiency associated with this finding was the failure of engineering personnel to correct a condition adverse to quality. **The finding is greater than minor because it was associated with the design control performance attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the reliability and availability of systems that respond to initiating events**. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because there was no actual loss of safety function. This issue involved problem identification and resolution crosscutting aspects associated with engineering personnel implementing timely corrective actions.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality, are promptly identified and corrected. Contrary to this, in April 2003 the licensee identified the need to perform a LDCR and a corresponding 10 CFR 50.59 screening due to abandonment of the UFSAR-required boronometer, but failed to implement corrective actions to ensure that the LDCR and 10 CFR 50.59 screening were performed. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2823704, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528; 05000529; 05000530/2005004-06, "Failure to Perform a Licensing Document Change Request and 10 CFR 50.59 Screening for Abandonment of the Boronometer."

.3 Resolution of Licensee Response to NRC Concerns

a. Inspection Scope

The inspectors reviewed the licensee's response to NRC concerns regarding the setting of bistable setpoints in accordance with Revision 4 and 5 of Design Calculation 13-JC-SE-201, "Ex-Core Safety Channel Log Power Instrument Setpoint and Uncertainty Calculation." This included a review of correspondence from the licensee, a detailed review of the above calculations with their associated field implementation procedures, review of information obtained by the NRC Office of Investigations in a related investigation (OI 4-2004-032), and interviews with station personnel.

b. Findings

1. Improper Control of Design Parameters

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, for the improper control of design parameters for the ex-core nuclear instrument safety channels.

Description. The inspectors determined that Design Calculation 13-JC-SE-201, "Ex-Core Safety Channel Log Power Instrument Setpoint and Uncertainty Calculation," Revision 4, did not adequately account for instrument uncertainty. As stated in Revision 4, the design setpoints were established as 3.699 vdc for Log 1 and 2 bistable settings. However, this corresponded to the same value as the TS limits (i.e., no explicit uncertainty). The inspectors also found that engineering personnel were aware that the Log 1 and 2 bistable settings in Revision 4 were non-conservative, but had addressed this in field implementation (maintenance) procedures instead of the design calculation. The field procedure called for establishing a 0.02 vdc offset from the design setting to account for the inaccuracy. The inspectors concluded that field maintenance procedures did not have the same reviews or controls as the design calculations required and, therefore, could have been changed without adequately protecting design limits. If the offsets were removed from the field procedures, or if maintenance personnel had requested administrative approval to apply already accepted design limits, the setpoints would have been set non-conservatively.

Analysis. The deficiency associated with this finding was a failure of engineering personnel to maintain design basis information. **The finding is greater than minor because if left uncorrected it could become a more significant safety concern in that failures to maintain design calculations could result in the incorrect setting of safety related devices. This finding is associated with the design control performance attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the reliability and availability of systems that respond to initiating events.** Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because there was no actual loss of safety function.

Enforcement. Criterion III of Appendix B to 10 CFR Part 50 states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to this, engineering personnel did not correctly translate design requirements, nor did they properly control design basis information regarding ex-core safety channels. Specifically, TS required values were being maintained apart from design calculations and documents. Because the finding is of very low safety significance, has been entered into the licensee's CAP as

CRDR 2612092, and the licensee corrected the condition with Revision 5 to the design calculation, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528-529-530/2005005-07, “Improper Control of Design Parameters for the Ex-Core Safety Channels.”

2. Incomplete and Inaccurate Information Provided to the NRC

Introduction. The inspectors identified a noncited Severity Level IV violation of 10 CFR 50.9 for providing incomplete or inaccurate information to the NRC. The underlying safety significance of the violation was very low.

Description. On June 27, 2003, the licensee provided the NRC with a letter (102-04958-GRO/SAB/DJS) regarding whether or not there were errors in the TS required setpoints for the Log 1 and 2 bistables, and whether these settings were properly addressed in design documents to ensure protection of the TS limits. The licensee responded in a manner in which it was described that the setpoints were adequately conservative and that there was sufficient margin in the settings. It is common for licensees to employ an implicit method for instrument uncertainties that addresses these values in the available margin of the calculation. Based upon the licensee’s response, NRC inspectors initially concluded that the licensee appropriately answered the safety concern. However, based upon a followup review of Revision 5 to this calculation, approved in June of 2004, the inspectors found there was no available margin and the assurance of TS limits needed to be addressed with an explicit 0.02 vdc offset in the design calculation. Inspectors also found that engineering personnel had been addressing this error in another revision to the calculation since some time in 2001 (i.e., prior to response to the NRC’s concerns).

The inspectors conducted several interviews with engineering personnel and other station personnel regarding Design Calculation 13-JC-SE-201, Revisions 4 and 5. The NRC also conducted interviews with the assistance of the Office of Investigations. One of the individuals interviewed was the engineer responsible for the instrument setpoint methodology and program. During the interviews, NRC inspectors were able to determine that certain facts and conclusions previously provided by the licensee, were not representative of the actual process of establishing and maintaining the Log 1 and 2 bistable setpoints in accordance with Design Calculation 13-JC-SE-201. For example, the inspectors determined that the NRC needed two other procedures to follow the methodology for setpoints, neither of which were provided with the licensee’s earlier responses. The inspectors determined that regardless of the format of the methodology, the control of design setpoints was not in accordance with NRC requirements, and that had the licensee provided complete information about Revision 5, the NRC would have come to this conclusion earlier.

Analysis. This finding was not assessed via the significance determination process of NRC Manual Chapter 0609 because the licensee’s actions impeded the regulatory process. Therefore, this finding was assessed in accordance with the NRC Enforcement Policy. The inspectors determined that engineering personnel had additional information, including the subsequently corrected revision of the calculation going through final verification, and additional explanatory setpoint procedures, which were not referenced or provided during the original correspondence by the licensee. Had complete and accurate information been supplied at the time of the original request in 2003, the NRC would have identified a design control violation at that time (Section 4OA2.3.1). The safety consequence of this issue are the same as the underlying technical issue, very low safety significance (Green), in that there was no actual loss of safety function.

Enforcement. 10 CFR 50.9 requires that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. Contrary to this, on June 27, 2003, the licensee provided incomplete and inaccurate information regarding the design control of ex-core safety channel log power instrument setpoints. This information was determined to be material in that it affected the NRC's ability to determine compliance with NRC requirements. Because of the very low safety significance and because the licensee entered this issue into the CAP as CRDR 2829051, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528-529-530/2005005-08, "Incomplete and Inaccurate Information Associated with the Ex-Core Safety Channels."

.4 Crosscutting Issues Followup Inspections

The inspectors reviewed CRDRs 2780273 and 2780286, which documented the NRC's identification of substantive crosscutting issues in the human performance and problem identification and resolution areas, respectively. The substantive crosscutting issues were initially documented in the NRC's annual assessment letter to Palo Verde dated March 2, 2005. Continuance of the substantive crosscutting issues was documented in the NRC's mid-cycle assessment letter to Palo verde dated August 30, 2005. The inspectors conducted periodic discussions with licensee management to monitor their progress in addressing the substantive crosscutting concerns. The licensee initiated CRDR 2822493 which supported the development of the Performance Improvement Plan as an addendum to the Business Plan. As of the end of the inspection period, the licensee's corrective actions for the substantive crosscutting issues had not been completed. As highlighted in Sections 4OA2.5 and 4OA4, several additional crosscutting issues were identified during the inspection period. These examples indicate that the licensee's corrective actions in response to the substantive crosscutting issues have not reduced the frequency of human performance and problem identification and resolution issues.

.5 Cross-References to Problem Identification and Resolution Findings Documented Elsewhere

Section 1R14.1 describes a finding that involved the failure to correct conditions adverse to quality.

Section 1R19.2 describe findings that involved the failure to identify conditions adverse to quality.

Section 4OA2 describe findings that involved the failure to implement timely corrective actions.

4OA3 Event Followup (71153, 71111.14)

.1 (Closed) Licensee Event Report (LER) 05000528/2004008-00, "Improper Contact Configuration on Containment Isolation Valve"

On June 10, 2004, while performing routine status display checks, a control room reactor operator noticed that a light on the Safety Equipment Actuation Status for hydrogen recombiner outboard containment isolation Valve 1JHPBUV004 was illuminated. The operator questioned the indication since the valve was in the closed position as indicated on the valve's handswitch and the emergency response facility data acquisition display system computer. After verifying the valve was closed, a WO was generated to troubleshoot and correct the indication problem. On July 19, 2004, troubleshooting efforts identified that a rotor cam was oriented such that the limit switch in this bank would actuate opposite of the desired control logic for the safety equipment actuation status and Containment Isolation Actuation Signal (CIAS). The licensee concluded that the limit switch was improperly set on May 24, 2004, due to a maintenance personnel error while performing WO 2571040. With Valve 1JHPBUV004 in the open position, the valve's CIAS closing circuit was interrupted which would prevent the valve from closing upon a CIAS. The surveillance test used to determine operability of the valve following the maintenance only verified the closure time of the valve and did not verify the CIAS function required by the motor operator valve program.

This finding is determined to be greater than minor because it was associated with the configuration control attribute of the containment barrier integrity cornerstone and affects the cornerstone objective of preserving the containment boundary physical design to protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding did not represent an actual open pathway in the physical integrity of reactor containment. This licensee identified finding involved a violation of TS 5.4.1.a. The enforcement aspects of the violation are discussed in Section 4OA7. The LER is closed.

.2 (Closed) LER 05000528,05000529;05000530/2004011-00, "Missed Surveillance Requirements for Containment IV's Test & Drain Valves"

This issue was previously dispositioned as NCV 05000528/2004005-04; 05000529/2004005-04; and 05000530/2004005-4, "Failure to Include Vents and Drains into Locked Valve Program." The inspectors reviewed the LER and found no additional concerns. This LER is closed.

.3 (Closed) LER 05000528/2005003-00, “Calibration Method That Might Have Failed to Provide Reactor Protection During Low Power Operation”

On March 24, 2005, a concern regarding the Excore Log Safety Channels was raised about whether the channels were linear over their entire span. The channels were calibrated at essentially full power, which is in the mean square voltage region. However, the safety function is of concern at low powers in the log count rate region. The log count rate and mean square voltage regions did not adequately align such that the calibration method used by the licensee resulted in one channel not being able to provide reactor protection during very low power operation. Corrective actions have been implemented by the licensee to ensure that there is no loss of safety function. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the problem in CRDR 2760452. This LER is closed.

.4 (Closed) LER 05000529/2004002-00, “Reactor Trip on Low DNBR”

On July 14, 2004, the licensee was in a severe thunderstorm in which several lightning strikes landed in the vicinity of the Palo Verde switchyard. After a series of lightning strikes, the Unit 2 main generator tripped causing a reactor power cutback. Approximately 11 seconds later, a low departure from nucleate boiling ratio reactor trip was initiated by the CPCs. The inspectors reviewed CRDR 2721635 and its significant root cause investigation. The licensee concluded that the lightning strikes damaged the main generator excitation and voltage regulation system causing the generator trip. The reactor trip that followed was caused by a conservative planar radial peaking factor in the CPCs, which produced a conservative departure from nucleate boiling value greater than the trip setpoint that existed in the reactor. The licensee replaced the failed excitation and voltage regulation system and revised the planar radial peaking factor in the CPCs to minimize the possibility of another automatic reactor trip following a reactor power cutback. No new findings were identified in the inspector’s review. This LER is closed.

.5 (Closed) LERs 05000528/2004005-00 and 2004005-01, “Missed ST on Shutdown Cooling Valve RCS Pressure Interlocks”

On May 28, 2004, while revising a surveillance procedure, licensee personnel determined that Palo Verde was not testing shutdown cooling isolation valve interlocks as described in UFSAR Section 7.6, nor were these safetyrelated interlocks being subjected to a complete channel calibration and functional test as described in NRC Generic Letter 96-01, “Testing of Safety-Related Circuits.” The licensee determined that during a revision in 1995, sections of the procedure that were required by the UFSAR were removed. The significant root cause investigation attributed the deletion to human error and an improper 10 CFR 50.59 screening. On June 30, 2004 the licensee revised the surveillance procedure and tested all three units. The licensee then reported to the NRC on July 23, 2004, that all testing in accordance with the UFSAR had been completed. On September 25, 2004, the licensee determined that their corrective actions were not adequate and that the surveillance procedure was still not in compliance with the UFSAR. The surveillance procedure was revised and implemented again. On August 2, 2005, the licensee determined that their corrective actions were not adequate in that testing was not completed for Unit 1. The licensee finally completed all required testing in accordance with the UFSAR on August 14, 2005.

The failure to implement appropriate corrective actions represented a finding that is greater than minor, since the failure to correct conditions adverse to quality could become a more significant safety concern. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because there was no loss of system safety function and when the surveillance procedure was implemented, the system passed all the required testing. This licensee-identified finding involved a violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action.” The enforcement aspects of this violation are discussed in Section 4OA7.

Reporting inaccurate information to the NRC regarding the completion of the testing is a finding that was not assessed via the significance determination process of NRC Manual Chapter 0609 because the licensee’s actions impeded the regulatory process. Therefore, this finding was assessed in accordance with the NRC Enforcement Policy. The inspectors determined that the safety consequences of this issue would be the same as the corrective action issue, very low safety significance (Green), in that there was no actual loss of a safety function. This licensee-identified finding involved a violation of 10 CFR 50.9, “Completeness and Accuracy of Information.” The enforcement aspects of this violation are discussed in Section 4OA7. This LER is closed.

4OA4 Crosscutting Aspects of Findings

Section 1R14.1 describes a finding where inadequate procedure implementation and questioning attitude resulted in SFP leakage to the environment.

Section 1R14.2 describes a finding where inadequate procedure implementation and questioning attitude resulted in a reactor trip and main steam isolation.

Section 1R15 describes a finding where poor work controls resulted from inadequate communications between the engineering, maintenance, and operations organizations .

Section 1R19.1 describes a finding where inadequate implementation of work order requirements and a lack of attention to details resulted in failing to promptly address potential equipment deficiencies.

Section 1R19.2 describes a finding involving inadequate implementation of a procedure when performing maintenance and testing on safety related equipment.

4OA6 Meetings, Including Exit

On October 4, 2005, the resident inspectors presented the inspection results of the resident inspections to Mr. J. Levine, Executive Vice President, Generation, and other members of the licensee management staff. The licensee acknowledged the findings presented. The inspectors noted that while proprietary information was reviewed, none would be included in this report.

On September 13, 2005, the Engineering Branch Chief presented the results of a followup inspection in an area of problem identification and resolution to Mr. J. Levine, Executive Vice President, Generation, and other members of licensee management. The licensee acknowledged the findings presented.

On September 14, 2005, the emergency preparedness inspector conducted a telephonic exit meeting to present the inspection results to Mr. E. O'Neil, Department Leader, Emergency Preparedness, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- 10 CFR Part 50, Appendix B, Criterion VI, "Document Control," requires, in part, that measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed.

  Contrary to this requirement, an April 2005 Audit Report performed by the licensee identified broad shortcomings in their management of quality assurance records, including control room drawings not receiving required updates, design change packages not being archived in a timely fashion, and configuration databases not being updated after completed design changes. The Audit Report documented thousands of occurrences of these errors. As a result, the licensee initiated seven CRDRs to correct weaknesses in the Quality Assurance Program. This finding is greater than minor because if left uncorrected it could become a more significant safety concern in that errors in design documents could result in misoperation of plant equipment. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because there was not a loss of safety function.

- 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances.

  On July 26, 2005, the licensee identified a deficiency with Procedure 73ST-9SG05, “ADV Nitrogen Accumulator Drop Test,” Revision 20. This procedure is used to satisfy TS 3.7.4 and TS 5.5.8 by leak testing several pressure supply and check valves in the pneumatic control portion of the atmospheric dump valves (ADVs). The test pressurizes the pneumatic control portion of the ADV with nitrogen from the accumulators and measures the pressure drop over a two hour period. If there is leakage in the system the pressure drop will be excessive and the ADV would be declared inoperable. Engineering personnel raised the concern that the ADV used more nitrogen when open, and therefore, should be tested in the open position. Further analysis of the testing methodology on July 27, 2005, identified a portion of the pneumatic control system that was isolated when the ADV was in the closed position and therefore not tested. The licensee concluded that the testing methodology was inadequate and revised Procedure 73ST-9SG05, “ADV Nitrogen Accumulator Drop Test,” to include testing of the ADVs in the open and closed positions. This issue is documented in the licensee’s CAP as CRDR 2818612. This finding is of very low safety significance because there was no actual loss of safety function when the ADVs were appropriately tested.

- Technical Specification 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 9.a, requires procedures for performing maintenance that can affect the performance of safety related equipment. Contrary to this, on May 24, 2004, operations personnel did not follow the instructions in WO 2571040, which contained instructions on the proper setting of the limit switch for the hydrogen recombiner outboard containment isolation Valve 1JHPBUV004. This finding was documented in CRDR 2722547 and LER 05000528/2004008-00 (see Section 4OA3.1).

- 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to this, the licensee failed to implement adequate corrective actions after determining that testing of the shutdown cooling isolation valve interlocks was not performed as described in the UFSAR. This finding was documented in CRDR 2687507 and LERs 05000528/2004005-00 and 2004005-01 (see Section 4OA3.5).

- 10 CFR 50.9, “Completeness and Accuracy of Information,” requires, in part, that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. Contrary to this, on July 23, 2004, the licensee incorrectly reported to the NRC that an inadequate surveillance procedure used to test the shutdown cooling isolation valve interlocks had been revised and all units had been tested. On September 25, 2004, the licensee determined that their corrective actions were not adequate and that the surveillance procedure was not in compliance with the UFSAR, as reported to the NRC. The licensee then revised the procedure and successfully tested all three units. This finding was documented in CRDR 2819921 and LERs 05000528/2004005-00 and 2004005-01 (see Section 4OA3.5).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Hautala, Senior Compliance Engineer
E. O'Neil, Department Leader, Emergency Preparedness
S. Bauer, Department Leader, Regulatory Affairs
P. Borchert, Director, Work Management
R. Buzard, Senior, Consultant, Regulatory Affairs
D. Carnes, Director, Nuclear Assurance
P. Carpenter, Unit Department Leader, Operations
C. Churchman, Director, Engineering
S. Coppock, Department Leader, System Engineering
D. Fan, Department Leader, Design Mechanical Engineering
J. Gaffney, Director, Radiation Protection
J. Hesser, Director, Emergency Services
P. Kirker, Unit Department Leader, Operations
D. Marks, Section Leader, Regulatory Affairs - Compliance
D. Mauldin, Vice President, Engineering and Support
M. McGhee, Unit Department Leader, Operations
M. Muhs, Department Leader, Maintenance
M. Radsprinner, Section Leader, Systems Engineering
T. Radtke, Director, Operations
F. Riedel, Director, Nuclear Training Department
J. Scott, Section Leader, Nuclear Assurance
C. Seaman, Director, Regulatory Affairs
M. Shea, Director, Maintenance
D. Smith, Plant Manager, Production
M. Sontag, Department Leader, Nuclear Assurance
D. Straka, Senior Consultant, Regulatory Affairs
R. Stroud, Senior Consultant, Regulations Affairs
J. Taylor, Department Leader, Operations Support
T. Weber, Section Leader, Regulatory Affairs

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000528/2005004-01	NCV	Failure to Monitor Telltale Drains Resulted in Spent Fuel Pool Leakage to the Environment (Section 1R14.1)
05000528/2005004-02	NCV	Improper Control of Steam Generator Feedwater System Resulted in a Reactor Trip and Main Steam Isolation (Section 1R14.2)

05000528/2005004-03	FIN	Communication Deficiencies Between Organizations (Section 1R15)
05000529/2005004-04	NCV	Improper Visual Analysis of Bearing Oil Sample (Section 1R19.1)
05000529/2005004-05	NCV	Failure to Follow Procedures During Core Protection Calculator Software Installation and Testing (Section 1R19.2)
05000528; 05000529; 05000530/2005004-06	NCV	Failure to Perform a Licensing Document Change Request and 10 CFR 50.59 Screening for Abandonment of the Boronometer (Section 4OA2.2)
05000528-529-530/2005004-07	NCV	Improper Control of Design Parameters for the Ex-Core Safety Channels (Section 4OA2.3.1)
05000528-529-530/2005004-8	NCV	Incomplete and Inaccurate Information Associated with the Ex-Core Safety Channels (Section 4OA2.3.2)

Closed

05000528/2004008-00	LER	Improper Contact Configuration on Containment Isolation Valve (Section 4OA3.1)
05000528,05000529; 05000530/2004011-00	LER	Missed Surveillance Requirements for Containment IV's Test & Drain Valves (Section 4OA3.2)
05000528/2005003-00	LER	Calibration Method That Might Have Failed to Provide Reactor Protection During Low Power Operation (Section 4OA3.3)
05000529/2004002-00	LER	Reactor Trip on Low DNBR (Section 4OA3.4)
05000528/2004005-00	LER	Missed ST on Shutdown Cooling Valve RCS Pressure Interlocks (Section 4OA3.5)
05000528/2004005-01	LER	Missed ST on Shutdown Cooling Valve RCS Pressure Interlocks (Section 4OA3.5)

Discussed

None

LIST OF DOCUMENTS REVIEWED

In addition to the documents called out in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Section 1R01: Adverse Weather Protection

CRDR
CRAI 2806021

Miscellaneous
Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," Inspection Report 50-346/2004-017, Operability Determination 285, "Water Intrusion into Diesel Fuel Oil Storage Tank Vault"

Section 1R04: Equipment Aligment

Miscellaneous
Specification 13-CN-380, "Installation Specification for Seismic Category IX Scaffolding," Revision 4

Drawings
01-M-DGP-001, "P&I Diagram Diesel Generator System," Sheet 1 of 9, Revision 45
01-M-DGP-001, "P&I Diagram Diesel Generator System," Sheet 2, Revision 45
01-M-DGP-001, "P&I Diagram Lube Oil Diesel Generator System," Sheet 3, Revision 45
01-M-DGP-001, "P&I Diagram Jacket Water Diesel Generator System," Sheet 4, Revision 45
01-M-DGP-001, "P&I Diagram Diesel Generator System," Sheet 9, Revision 45

Section 1R05: Fire Protection

Miscellaneous
Pre-Fire Strategy

Section 1R06: Flood Protection Measures

CRDRs
2-8-0010, 2548036, 2746319, 2616959, 2797521, 2809519

Work Orders
2745724, 2745728, 2745730, 2794590

Section 1R12: Maintenance Implementation

Miscellaneous
System Health Report

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
30DP-9MT03, “Assessment and Management of Risk When Performing Maintenance in Modes 1 - 4,” Revision 11

73ST-9AF02, “AFA-P01 - Inservice Test,” Revision 33

73ST-9XI38, “AF Pumps Discharge Check Valves - Inservice Test,” Revision 12

CRDRs
2818957, 2635507, 1-8-0280

Drawings
01-P-SIF-105, “Containment Building Isometric Safety Injection System Shutdown Cooling Lines,” Sheet 1 of 2

Miscellaneous
Engineering White Paper - U1 SIAV522 Leakage
Engineering Evaluation Request 89-SI-340

Section 1R14: Operator Performance During Nonroutine Plant Evolutions and Events

Drawings
13-J-SAS-001, “Engineering Safety Features Actuation System (ESFAS) Train A Actuated Devices,” Sheet 2

N001-1306-162, “ESFAS Auxiliary Relay Cabinets Electrical Schematics,” Revision 5

13-C-ZFS-420, “Fuel Building Liner Plate Sections and Details,” Revision 11

13-P-ZFE-504, “Fuel Building Plumbing Details,” Revision 4

01-M-RDP-005, “P&I Diagram, Radioactive Waste Drain System (Fuel Building),” Revision 8

13-C-ZFS-400, “Fuel Building Liner Plate Supporting Steel,” Revision 10

13-C-ZFS-402, “Fuel Building Liner Plate Elevations Sheet 1,” Revision 10

13-C-ZFS-180, “Fuel Building Concrete Sections and Details Sheet 1,” Revision 8

13-C-ZFS-186, “Fuel Building Portal Plans Below ED 100'-0" Sections and Details,” Revision 5

13-C-ZFS-100, “Fuel Building Area F1A & F1B Plan at E1 100'-0," Revision 17

13-C-ZFS-130, “Fuel Building Area F3A & F3B Plan at E1 140'-0," Revision 14

13-C-ZFS-401, “Fuel Building Liner Plate Bottom Panels,” Revision 7

13-C-ZFS-421, "Fuel Building Liner Plate Sections and Details," Sheet 2, Revision 6

13-C-ZFS-204, "Fuel Building Fuel Reconstitution Platform Plans," Sections and Details Sheet 2, Revision 1

CRDRs
2872981, 2813049, 2813074, 2813076, 2327873, 2597895, 2814209, 2827635, 2815935

Section 1R15: Operability Evaluation

Procedures
73ST-9SG05, "ADV Nitrogen Accumulator Drop Test," Revision 20
40DP-90OP02, "Conduct of Shift Operations," Revision 31
33MT-9ZZ02, "Freeze Sealing," Revisions 0 and 5
93DP-0LC07," 10 CFR 50.59 and 72.48, "Screenings and Evaluations," Revision 8
40DP-9OP26, "Operability Determination," Revision 13

Drawings
933900051, "ADV Control Schematic"
01-J-SGE-001, "Pneumatic Loop Diagram ADVs," Revision 5
01-M-SGP-001, "Main Steam System," Revision 50

Work Order
2813864

Miscellaneous
Structural Integrity Associates letter to APS, SIR-05-221, dated July 12, 2005

SIR-05-221, Revision 1, dated July 12, 2005

Ultrasonic Thickness Examination Reports 05-316, 05-318

Calculation 13-MC-SG-0314, "Nitrogen Tank Pressure Requirements for ADVs," Revision 5

Palo Verde Engineering - 2005 Strategic Plan

10 CFR 50.59 Resource Manual, Revision 2, April 2003

10 CFR 50.59 Review for Procedure 30MT-9ZZ02

Specification 13-PN-204, "Fabrication and Installation of Nuclear Piping Systems for the Arizona Public Service Company PVNGS Units 1, 2, and 3," Revision 11

NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, November 2000

CRDRs
2813750, 2820807, 2820126

Section 1R16: Operator Workarounds

Miscellaneous
Operator Boards Printout
Operations Challenges Tracking Form

Section 1R19: Post Maintenance Testing Checklist

Work Orders
2820466, DFWO 2824743, DFWO 2824397

CRDRs
2824255, 2824258, 2825189, 2824269, 2821341

Miscellaneous
Westinghouse CAPS Issue Report 05-138-W008, "Two Process Input Channel Failures in CPC," Revision 3

Doc ID JN1000-A00109, Revision 6

Software release memo for common Q based CPC system

ERFDADS printouts of SI flowrates on August 15, 2005

ASME OM Code 2001, Subsection ISTC-5220, "Check Valves"

10 CFR 50.59 Screening S-05-0272, Revision 0

Procedures
73ST-9DG02, "Class 1E Diesel Generator and Integrated Safeguards Test Train B," Revision 11

40ST-9DG02, "Diesel Generator B Test," Revision 26

73ST-9ZZ25, "Check Valve Disassembly, Inspection, and Manual Exercise," Revision 5

73ST-9XI33, "HPSI Pump and Check Valve Full Flow Test," Revision 32, Appendix D

40OP-9SI02, "Recovery from Shutdown Cooling to Normal Operating Lineup," Revision 60

Section 1R20: Refueling and Other Outage Activities

Miscellaneous
Permit 119092, "S1A-V522 Intrusive Rework"

Section 1R22: Surveillance Testing

Procedures

73ST-9SG05, "ADV Nitrogen Accumulator Drop Test," Revisions 20 and 22

CRDRs
2820807, 2818612, 2818836

Drawings
01-J-SGE-001, Revision 5
933900051, Code 19562
01-M-SGP-001, Revision 50

Miscellaneous
Calculation 13-MC-SG-0314

Section 1R23: Temporary Plant Modifications

Miscellaneous
740P-9SP02, ESPS Chemical Addition T-Mod Installation, Operation, and Removal," Revision 2
Unit 2 temporary log sheet, dated September 16, 2005
73DP-9CY05, "Systems Chemistry Specification," Revision 31
93DP-OLC03, "Licensing Document Maintenance," Revision 13
50.59 evaluations 01-00051 and 99-00157

Section 4OA2: Identification and Resolution of Problems

Procedures
36MT-9CH01, "Boronometer Calibration," Revision 5
72ST-9RX01, "Core Reactivity Balance," Revision 9
74OP-9SS01, "Primary Sampling Instructions," Revision 28
74CH-9ZZ06, "Boron Autotitrator Operation and Calibration," Revision 25
90DP-0IP10, "Condition Reporting," Revisions 15 and 19
96DP-0LC07, "10 CFR 50.59 and 72.48 Screenings and Evaluations," Revision 8

Section 40A7: Licensee-Identified Violations

CRDR's
2598433, 2712596, 2784544, 2785708, 2786615, 2786616, 2786619, 2786620, 2786621, and 2792404

Miscellaneous
PVNGS Audit Report 05-005, "Design Control"
Regulatory Guide 1.64, Revision 2, June 1976, "Quality Assurance Requirements for the Design of Nuclear Power Plants"
UFSAR, Chapter 17.2, "Quality Assurance During the Operations Phase"

Procedures
81DP-0EE10, "Plant Modifications," Revision 11
81DP-0DC13, "Deficiency (DF) Work Order," Revision 15
81DP-0DC16, "Engineering Document Change (EDC)," Revision 15
84DP-0RM30, "Document/Record Control and Turnover," Revision 16
81DP-0CC05, "Design and Technical Document Control," Revision 27
90DP-0IP10, "Condition Reporting," Revision 19

LIST OF ACRONYMS

ADV	atmospheric dump valve
CAP	corrective action program
CFR	*Code of Federal Regulations*
CIAS	containment isolation actuation signal
CPC	core protection calculator
CRAI	condition report action item
CRDR	condition report/disposition request
CRS	control room supervisor
HPSI	high pressure safety injection
I&C	instrument and control
LDCR	licensing document change request
LER	licensee event report
NCV	noncited violation
NRC	Nuclear Regulatory Commission
OD	operability determination
RCP	reactor coolant pump
RCS	reactor coolant system
SFP	spent fuel pool
SG	steam generator
SSC	structures, systems, and components
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WO	work order