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

August 2, 2005

Gregg R. Overbeck, Senior Vice
President, Nuclear
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000528/2005003, 05000529/2005003, AND 05000530/2005003

Dear Mr. Overbeck:

On June 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 June 28, 2005, with Mr. Mauldin 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 three NRC identified findings and five self-revealing findings. Four of these findings were evaluated under the risk significance determination process as having very low safety significance (Green). Four findings were not suitable for evaluation under the significance determination process; however, they were determined to be of very low safety significance (Green) by NRC management review. Seven of these 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. 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 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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Dockets: 50-528
50-529
50-530
Licenses: NPF-41
NPF-51
NPF-74

Enclosure:
NRC Inspection Report 05000528/2005003,
05000529/2005003, and 05000530/2005003
w/Attachment: Supplemental Information

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SISP Review Completed: _TWP__ ADAMS: / Yes ☐ No Initials: __TWP_
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| RIV:RI:DRP/D | RE:DRP/D | RI:DRP/D | SRI:DRP/D | C:DRS/OB |
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| PLBenvenuto | SMAlfernik | JFMelfi | GGWarnick | ATGody |
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| C:DRS/PSB | C:DRS/EMB | C:DRS/PEB | C:DRP/D | | |
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| MPShannon | JAClark | LJSmith | TWPruett | | |
| **/RA/** | **/RA/** | **GDReplogle** | **/RA/** | | |
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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: | 05000528/2005003, 05000529/2005003, 05000530/2005003 |
| Licensee: | Arizona Public Service Company |
| Facility: | Palo Verde Nuclear Generating Station, Units 1, 2, and 3 |
| Location: | 5951 S. Wintersburg Road<br>Tonopah, Arizona |
| Dates: | April 1 through June 30, 2005 |
| Inspectors: | G. Warnick, Senior Resident Inspector, Project Branch D<br>J. Melfi, Resident Inspector, Project Branch D<br>P. Benvenuto, Resident Inspector, Project Branch D<br>S. Alferink, Reactor Inspector<br>P. Elkmann, Emergency Preparedness Inspector<br>L. Carson II, Senior Health Physicist<br>W. Sifre, Reactor Inspector, Engineering Branch<br>E. Owen, Reactor Inspector, Engineering Branch<br>N. Taylor, Project Engineer, Project Branch D<br>G. Werner, Senior Project Engineer, Project Branch D |
| Approved By: | Troy W. Pruett, Chief, Project Branch D<br>Division of Reactor Projects |

Enclosure

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## SUMMARY OF FINDINGS

IR 05000528/2005003, 05000529/2005003; 05000530/2005003; 04/01/05 - 06/30/05; Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Integrated Resident and Regional Report; Equipment Alignment, Maintenance Effectiveness, Nonroutine Evolutions, Operability Evaluations, Operator Workarounds, and Surveillance Testing.

This report covered a 3-month period of inspection by three resident inspectors, three reactor inspectors, one emergency preparedness inspector, one senior health physicist, and two project engineers. 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 for the failure to follow procedures which resulted in an inadvertent reduction of spent fuel pool water level. Specifically, approximately 1800 gallons of water was unknowingly directed to the transfer canal when operations personnel failed to follow Procedure 40OP-9PC06, "Fuel Pool Clean Up and Transfer." The initial auxiliary operator opened a valve when the step required the valve to be closed and did not open another valve as required by the procedure. A second auxiliary operator performed an inadequate independent verification of the position of the valves. This issue involved human performance crosscutting aspects associated with procedure implementation and operator attention to detail. This issue was entered into the corrective action program as Condition Report/Disposition Request 2793816.

  The finding is greater than minor because it affects the configuration control 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 (Section 1R14).

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for an inadequate surveillance procedure which resulted in an inadvertent safety injection and subsequent reactor coolant system level transient. Specifically, an integrated safeguards test procedure cautioned operations personnel to evaluate the pressure difference between the reactor coolant system and safety injection tanks prior to any actuation that opened the safety injection tank outlet isolation valves. The procedure was inadequate in that it failed to caution the operator to consider level differences which could potentially impact the total pressure head of the system. This issue involved human performance crosscutting aspects associated with inadequate operations procedures. This issue was entered into the corrective action program as Condition Report/Disposition Request 2786378.

The finding is determined to be greater than minor because it affected the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Using Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” this finding is determined to have very low safety significance because the event did not constitute a loss of level control and did not represent a finding requiring quantitative assessment. The finding did not increase the likelihood of loss or cause a degradation in the ability to restore decay heat removal, reactor coolant system inventory, offsite power, alternate core cooling, or containment (Section 1R22).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to identify and correct a deficiency in the method of testing the auxiliary feedwater pump discharge check valves. Specifically, in 1998 the licensee identified the need to test the auxiliary feedwater pump Train B discharge check valve for leak tightness, but failed to implement the appropriate corrective actions to incorporate testing into Procedure 73ST-9XI38, “AF Pumps Discharge Check Valves - Inservice Test.” This issue involved problem identification and resolution crosscutting aspects associated with the failure to implement timely corrective actions. This issue was entered into the corrective action program as Condition Report/Disposition Request 2800972.

  The finding is greater than minor because it was associated with the equipment 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 1R04).

- Green. A self-revealing finding was identified for the failure to properly sequence work to maintain power to engineered safety features system cabinet Train B. Specifically, operations personnel prematurely implemented a tagout permit prior to restoring the redundant power supply following maintenance. The work sequencing performance deficiency resulted in the loss of vital power to the cabinet; thereby, initiating an inadvertent engineered safety features actuation. This issue involved human performance crosscutting aspects associated with inadequate communications between work control groups and a poor awareness of the plant configuration. This issue was entered into the corrective action program as Condition Report/Disposition Request 2796508.

The finding is greater than minor since it was associated with the configuration control attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the reliability and availability of systems that respond to initiating events. This finding cannot be evaluated by the significance determination process because Manual Chapter 0609, "Significance Determination Process," Appendix G, "Shutdown Operations Significance Determination Process," does not apply when the reactor is defueled. This finding is determined to be of very low safety significance by NRC management review because it was a deficiency that did not result in actual safety consequences since the reactor was defueled and a majority of the Train B equipment was tagged out for maintenance (Section 1R14).

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified for the failure to implement corrective actions to preclude repetition of a significant condition adverse to quality. Specifically, in 1988 the licensee identified that the gasket retaining bolts on several 16-inch butterfly valves were susceptible to stress corrosion cracking. The licensee only replaced bolts on the 16-inch valves with the identified failures and did not consider the need to replace bolts on similarly designed 10-inch and 24-inch valves. Consequently, in April 2005, the safety injection inboard and outboard containment sump isolation valves were discovered to have missing or degraded bolts, and the 10-inch containment spray to shut down cooling heat exchanger valves were determined to have suspect bolts. This issue involved problem identification and resolution crosscutting aspects associated with the failure to perform an adequate transportability review. This issue was entered into the corrective action program as Condition Report/Disposition Request 2791716.

  The finding is greater than minor since it affects the equipment performance attribute of the mitigating systems 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 1R15).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure to follow procedures to implement compensatory measures and properly track an operator workaround for a breaker handswitch with a broken operating knob. When questioned by the inspectors, operations personnel were not able to immediately locate the tools needed to operate the defective handswitch. This issue was entered into the corrective action program as Condition Report/Disposition Request 2807501.

  The finding is determined to be greater than minor because if left uncorrected it could become a more significant safety concern in that operators may not be able to operate equipment necessary to respond to initiating events. Using the Phase 1 Worksheet in

Manual Chapter 0609, "Significance Determination Process," the finding is determined to have very low safety significance because it only affects the mitigating systems cornerstone and did not result in the actual loss of a safety function (Section 1R16).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a for failing to ensure that maintenance on safety-related fuel handling equipment was performed by personnel with the correct qualifications. Specifically, the licensee failed to follow the requirements of Procedure 30DP-9MP01, "Conduct of Maintenance," which required that task-qualified independent workers (qualified workers) be assigned to perform work or direct work by dependent workers (unqualified workers). Consequently, maintenance on refueling equipment by unqualified workers had not been properly supervised on at least five work orders in 2003 and 2004. This issue was entered into the corrective action program as Condition Report/Disposition Request 2797536.

  The finding is determined to be greater than minor because if left uncorrected it could become a more significant safety concern in that improperly performed maintenance on fuel handling equipment could impact the safe movement of nuclear fuel and increase the probability of a fuel handling accident. 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 affects the barrier integrity cornerstone and is determined to be of very low safety significance by NRC management review because it was a deficiency that did not result in the actual degradation of spent fuel (Section 1R12).

- Green. Three examples of a self-revealing noncited violation of 10 CFR Part 50, Criterion V, “Instructions Procedures, and Drawings,” was identified for failing to properly implement procedures for refueling equipment. Specifically, refueling personnel did not: (1) complete a functional retest following maintenance on the spent fuel handling machine as required by Work Order 2781146; (2) ensure that spent fuel was in a safe condition, stop fuel handling operations, or contact the shift manager to determine the need to complete an event recovery checklist when a deficiency was identified with fuel handling equipment as required by Procedure 40DP-9OP02, “Conduct of Shift Operations”; and (3) ensure the material balance area short form was present on the spent fuel handling machine to perform proper independent verification or verify that the bridge and trolley were over the correct fuel assembly as required by Procedure 78OP-9FX03, “Spent Fuel Handling Machine.” This issue involved human performance crosscutting aspects associated with operator decision making and not following procedures. This issue also involved problem identification and resolution crosscutting aspects associated with the failure to correct a condition adverse to quality since there have been similar occurrences where operators failed to recognize the need to perform the event recovery checklist. This issue was entered into the corrective action program as Condition Report/Disposition Requests 2791974 and 2792326.

The finding is greater than minor since it could become a more significant safety concern if left uncorrected in that handling spent fuel with degraded equipment impacts the ability to safely handle spent fuel and increases the likelihood of a fuel handling accident. 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 affects the barrier integrity cornerstone and is determined to be of very low safety significance by NRC management review because it was a deficiency that did not result in the actual degradation of spent fuel (Section 1R14).

# REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full power for the entire inspection period.

Unit 2 operated at essentially full power until April 2, 2005, when the unit was shutdown for the twelfth refueling outage. The outage was completed on May 20, and the unit was returned to essentially full power on May 26 and remained there for the duration of the inspection period.

Unit 3 operated at essentially full power until May 23, 2005, when the unit was shut down to replace nine pressurizer heaters and to repair a reactor coolant pump oil leak. Following maintenance, the unit began heatup from Mode 4 on May 29, achieving Mode 2 on June 2. During the heatup, five additional pressurizer heater failures occurred and the unit returned to Mode 5 to replace the remaining 27 pressurizer heaters. Following replacement of the remaining heaters, the unit returned to full power on June 25 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)

Readiness for Seasonal Susceptibilities

a. Inspection Scope

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving extreme high temperatures. The inspectors (1) reviewed plant procedures, the Updated Final Safety Analysis Report, and Technical Specifications to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the below listed systems to ensure that adverse weather protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee would maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program (CAP) to determine if the licensee identified and corrected problems related to adverse weather conditions.

- June 28, 2005, Unit 1, spray pond system Trains A and B
- June 28, 2005, Unit 1 essential chilled water system Trains A and B
- June 28, 2005, Unit 1 Non class 13.8 Kv power

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the five 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 CAP to ensure problems were being identified and corrected.

C April 6, 2005, Unit 2, shutdown cooling system Train A during midloop operations

C April 20, 2005, Unit 2, fuel pool cooling system Trains A and B while both trains were running for spent fuel pool decay heat removal

C April 25, 2005, Unit 2, containment spray, low pressure safety injection, and high pressure safety injection system Train A

C April 26, 2005, Unit 2, emergency diesel generator system Train A while Train B was out of service for maintenance

C June 8, 2005, Unit 3, shutdown cooling system Train B

The inspectors completed five samples.

.2 Complete Walkdown

The inspectors: (1) reviewed plant procedures, drawings, the Updated Final Safety Analysis Report, Technical Specifications, and vendor manuals to determine the correct alignment of the system; (2) reviewed outstanding design issues, operator workarounds, and CAP documents to determine if open issues affected the functionality of the system; and (3) verified that the licensee was identifying and resolving equipment alignment problems.

C May 20, 2005, Units 1, 2, and 3, auxiliary feedwater (AFW) system Trains A and B

The inspectors completed one sample.

b. Findings

Introduction. A Green noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified by the inspectors for the failure to identify and correct a deficiency in the method of testing AFW pump discharge check valves.

Description. The inspectors determined that the licensee had identified that a single active failure vulnerability associated with backflow through AFW pump discharge check Valves AFA-V015 and AFB-V024 had been raised in 1980, which resulted in a plant modification to AFW system Trains A and B. The modification involved installing two additional check valves upstream and in series with Valves AFA-V015 and AFB-V024 to add redundancy to the system. Concerns regarding whether the additional check valves performed a safety function and with the testing adequacy of all discharge check valves were documented in 1998 via Condition Report/Disposition Request (CRDR) 9-8-1508. The CRDR evaluation determined that the redundant check valves did not perform a safety function and that only Valves AFA-V015 and AFB-V024 required testing to comply with in-service testing requirements. The licensee modified procedures to test Valve AFA-V015, yet failed to implement the changes for testing Valve AFB-V024. Additional test methodology questions regarding the discharge check valves were documented in 2003 via CRDR 2624895.

On March 9, 2005, during the performance of the AFW turbine-driven pump quarterly surveillance test, licensee personnel again raised the concern that a condition was established during system restoration from the surveillance test where flow could pass through discharge check Valve AFA-V015. With Valve AFA-V015 open, there is a potential for backflow from the opposite train upon an AFW actuation signal which could pressurize the suction piping of the turbine-driven pump and render it inoperable. This potential issue was entered into the CAP as CRDR 2781507. Due to the potential single active failure vulnerability, the licensee modified Procedure 73ST-9XI38 to ensure Valve AFA-V015 was tightly closed after both full flow and recirculation flow testing of the turbine-driven pump.

On May 17, 2005, the inspectors questioned why this concern did not also apply to the AFW pump Train B discharge check Valve AFB-V024, since the piping configuration is similar to AFW system Train A. After further analysis, the licensee concluded the same potential for single active failure vulnerability existed for Valve AFB-V024 and determined that Procedure 73ST-9XI38 should have been performed when flow passed through this valve as well. As a consequence of the finding, on May 20, the licensee modified Procedure 73ST-9XI38 to include testing of Valve AFB-V024.

Analysis. The deficiency associated with this finding was the failure to correct a condition adverse to quality. The finding is greater than minor because it was associated with the equipment 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 (Green) because there was no actual loss of safety function. This issue involved problem identification and resolution crosscutting aspects associated with the failure to implement 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 1998 the licensee identified the need to test Valve AFB-V024 for leak tightness, but failed to implement corrective actions to incorporate testing into Procedure 73ST-9XI38. Additionally, the licensee missed several opportunities to correct the deficiency. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2800972, this violation is being treated as an NCV consistent with Section VI.A of the Nuclear Regulator Commission (NRC) Enforcement Policy: NCV 05000528; 05000529; 05000530/2005003-01, "Failure to Correct a Condition Adverse to Quality."

1R05 Fire Protection (71111.05)

a. Inspection Scope

Routine Inspection

The inspectors walked down the seven plant areas listed below to assess the material condition of active and passive fire protection features, their operational lineup, and their operational effectiveness. 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; (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 (7) reviewed the CAP to determine if the licensee identified and corrected fire protection problems.

C April 13, 2005, Unit 2, containment building, all accessible elevations

C April 18-20, 2005, Unit 1, auxiliary building, all accessible elevations

C April 24, 2005, Unit 1, control building, all accessible elevations

C April 24, 2005, Unit 2, control building, all accessible elevations

C May 2, 2005, Unit 2, condensate storage tank transfer pump house

C May 22, 2005, Unit 2, fuel building, all accessible elevations

- May 31, 2005, Unit 3, containment building, reactor coolant pump lube oil collection system

The inspectors completed seven samples.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

The inspectors reviewed test data from the performance test on the heat exchanger listed below. The licensee analyzed the heat exchanger performance data using Procedure 73DP-9ZZ10, “Guidelines for Heat Exchanger Thermal Performance Analysis,” Revision 4. The inspectors verified that: (1) test acceptance criteria and results considered differences between testing and design conditions, (2) inspection results were appropriately categorized against acceptable pre-established acceptance criteria, (3) the frequency of testing or inspection was sufficient to detect degradation prior to loss of the heat removal function, and (4) the test results considered instrument uncertainties.

- April 2, 2005, Unit 2, essential cooling water system Train B per Procedure 70TI-9EW01, “Thermal Performance Testing of Essential Cooling Water Heat Exchangers,” Revision 4.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

a. Inspection Scope

The inspectors observed and reviewed seven surface examinations and 26 volumetric examinations. The seven surface examinations included four magnetic particle examinations and three liquid penetrant examinations. All but one of the volumetric examinations were ultrasonic examinations. The exception was one radiographic examination. The inspectors also observed the ultrasonic system calibration. Of the examinations reviewed, two of them contained recordable indications that were accepted for continued service.

During the review of these examinations, the inspectors verified that the correct nondestructive examination procedure was used, examinations and conditions were as specified in the procedures, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed the documentation and verified that accepted indications were reviewed and dispositioned in accordance with the appropriate American Society of Mechanical Engineers (ASME) Code specified acceptance standards. The nondestructive examination certifications of personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspectors.

The inspectors also reviewed two Class 2 welds and verified that the weld process and postweld examinations were performed in accordance with the ASME Code.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

The review of the reactor vessel upper head inspections was performed under Temporary Instruction (TI) 2515/150 and is documented in Section 4OA5.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors performed a review of the licensee's boric acid walkdown of the Unit 2 reactor containment as documented on March 23, 2005. The inspectors verified that the visual inspections emphasized locations where boric acid leaks can cause degradation to safety significant components. The inspectors also reviewed one condition report and associated work orders which documented the boric acid leaks identified during the walkdown.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

b. Inspection Scope

The inspection procedure specified, with respect to in situ pressure testing, performance of an assessment of in situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results. The inspectors did not observe in situ pressure testing because none was required based on a review of the data.

The inspectors selected and reviewed the acquisition technique sheets and their qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy had been identified and qualified through demonstration.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. The purpose of the assessment is to identify degradation mechanisms and for each mechanism to determine proper detection technique, determine number of tubes, establish structural limits, and establish flaw growth rates. This inspection was the first inspection following the first operating cycle for replacement steam generators. Flaws identified were compared to pre-operation inspection data.

The inspection procedure specified confirmation be made that the steam generator tube eddy-current test (ET) scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors review determined that the steam generator tube ET scope and expansion criteria were satisfied.

The inspection procedure also specified, if the licensee identified new degradation mechanisms, to verify that the licensee had fully enveloped the problem in an analysis and had taken appropriate corrective actions before plant startup. At the time of this inspection, no new degradation mechanisms had been identified.

The inspection procedure required confirmation that all areas of potential degradation were being inspected, especially areas which were known to represent potential ET challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation, including ET challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure further required that repair processes being used were approved in the Technical Specifications for use at the site. At the time of this inspection, the licensee had not performed or used the designated Technical Specification approved repair processes, thus there was no opportunity to observe implementation of any potential repairs (e.g., plugging operations). The inspectors also verified that none of the flawed tubes identified by the licensee required in situ pressure testing.

The inspection procedure also required confirmation that the Technical Specification plugging limit was being adhered to and determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors confirmed that the licensee was adhering to these specifications.

The inspection procedure stated, if steam generator leakage greater than 3 gallons per day was identified during operations or during postshutdown visual inspections of the tubesheet face, to assess whether the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the ET probes and equipment were qualified for the expected types of tube degradation and assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of all ET performed. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements. The inspector assessed the site-specific qualifications of the techniques being used. Finally, the inspection procedure specified the review of one to five samples of ET data if questions arose regarding the adequacy of ET data analyses. The inspectors did not identify any results where ET data analyses adequacy was questionable.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed ten inservice inspection-related condition reports issued during the current and past refueling outage, and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee’s CAP, including the adequacy of the technical resolutions.

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 critique. The training scenario involved several different failures, including a loss of electrical power to a cooling tower, degraded condenser vacuum, loss of nuclear cooling water, and a loss of coolant accident.

- June 16, 2005, Scenario SES 0-03-00, "(Degraded Electrical) Loss of Condenser Vacuum/Loss of NC/LOCA," 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 below listed maintenance activity 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, 10 CFR Part 50, Appendix B, and Technical Specifications.

- April 20 through May 4 2005, Units 1, 2 and 3, effectiveness of preventive maintenance activities for all accessible fuel handling machines through inspection of equipment material condition

The inspectors completed one sample.

b. Findings

Introduction. A Green noncited violation was identified by the inspectors for failing to ensure maintenance on safety-related fuel handling equipment was performed by qualified personnel.

Description. Prior to Refueling Outage 2R12, routine maintenance on refueling equipment was performed by the Fuel Services Group. Members of the Fuel Services Group were required to complete Job Qualification Card NMO01-XX049 in order to be a task-qualified independent worker for work on refueling equipment. Per Procedure 15DP-0TR69, "Training and Qualification Administration," Revision 10, workers who have not completed the required training are permitted to perform work as dependent workers under the supervision of an appropriately qualified independent worker. Procedure 30DP-9MP01, "Conduct of Maintenance," Revision 37, Paragraph 3.8.2, requires that references be included in the Work Activity Sheet or the Work Order (WO) identifying the independent worker who oversees activities by the dependent worker.

Based on a request from the inspectors regarding the training of personnel who perform maintenance on fuel handling equipment, the licensee reviewed WO 2571663 and determined that the work had been done by a dependent worker, but that neither the WO nor the Work Activity Sheet recorded the identity of the independent worker. The licensee was able to locate a qualified Independent Worker who was on shift at the time, but could not connect the individual to the specific WO either through paperwork or the individual’s recollection of the job, which occurred in October 2003.

In an attempt to identify the scope of the issue, the licensee selected six WOs from 2003 and 2004 for review. Four of the six WOs were at least partially completed by dependent workers. None of the four WOs contained documentation of the identity of the independent worker who oversaw the maintenance activity. Based on the results of this review, the licensee concluded that some WOs were completed by unqualified workers in the past without the independent work requirement being met.

The inspectors reviewed each of the discrepant WOs with the licensee to identify the scope of work and opportunities to discover maintenance-induced failures after the work had been performed. In every case, the equipment had received multiple demands and/or preoperational checks during subsequent outages. The equipment serviced in the five WOs was compared with a list of known refueling equipment challenges in 2004-2005 with no direct correlation discovered.

The inspectors noted that this was not the first documented observation of this discrepancy. In CRDR 113786 on December 27, 1999, a similar condition was identified, after which paragraph 3.8.2 was added to Procedure 30DP-9MP01, "Conduct of Maintenance," to require that references be included in the Work Activity Sheet or the WO identifying the independent worker who oversees activities by the dependent worker.

The licensee stated that the responsibility for completing maintenance on fuel handling equipment was shifted prior to Refueling Outage 2R12 from the Fuel Services group to the Maintenance Department. Based upon the results of the initial review, the inspectors evaluated six more recent work orders from Refueling Outage 2R12 for review: The WOs were completed by independent workers with the requisite training (General Plant Mechanic, Electrician, or I&C Technician).

<u>Analysis</u>. The performance deficiency associated with this finding was the failure to ensure that personnel performing work on safety-related fuel handling equipment had the required qualifications to perform their work. The finding is determined to be greater than minor because if left uncorrected it could become a more significant safety concern in that improperly performed maintenance on fuel handling equipment could impact the safe movement of nuclear fuel and increase the probability of a fuel handling accident. 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 (SFP). This finding affects the barrier integrity cornerstone and is determined to be of very low safety significance by NRC management review because it was a deficiency that did not result in the actual degradation of spent fuel.

Enforcement. 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 9A, requires maintenance that can affect safety-related equipment be properly preplanned and performed in accordance with written instructions appropriate to the circumstances. Paragraph 3.4.1 of Procedure 30DP-9MP01, "Conduct of Maintenance," Revision 37, required that task-qualified independent workers be assigned to perform work or direct work by dependent workers. Contrary to this, the licensee failed to ensure independent workers directed refueling equipment maintenance completed by dependent workers. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2797536, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2005003-02, "Maintenance Performed on Fuel Handling Equipment Without Proper Qualifications."

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 activity 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) that 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) that the licensee identified and corrected problems related to maintenance risk assessments.

- June 13, 2005, Unit 2, evaluation of the configuration of one core protection calculator channel in trip while performing monthly excore detector calibration Procedure 36ST-9SE02, "Surveillance Test of Core Protection Calculators," Revision 43

The inspectors completed one sample.

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.

- April 16, 2005, Unit 2, electrical short on containment spray pump Train B as described in CRDRs 2790388 and 2791249

- April 22, 2005, evaluation of 10 CFR 50.59 screening and revision of Procedure 72ST-9RX03, "DNBR/LHR/AZTILT/ASI with COLSS Out of Service," Revision 11, Appendix B

- May 17, 2005, Unit 2, Technical Specification Limiting Condition for Operation 3.5.3, "Emergency Core Cooling System," Condition A, entry when an operator identified that the low pressure safety injection pump Train A seal failed following replacement as described in CRDR 2800779

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Plant 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 had identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the nonroutine evolutions sampled.

- On April 19, 2005, Unit 2, during spent fuel handling machine (SFHM) operations, the up-limit switch did not actuate when the Durant counter reached the "zero" point. Licensee investigation identified that the limit switch calibration was affected during maintenance the previous day. The equipment problem was not identified following maintenance since a functional retest was not performed. This event was documented in CRDR 2791974.

- On April 21, 2005, Unit 2, during performance of Westinghouse fuel inspections, the incorrect fuel assembly was grappled in the SFP and moved to the visual inspection stand where the error was identified upon visual verification of the fuel assembly serial number. This event was documented in CRDR 2792326.

- On April 23, 2005, Unit 1, while performing a routine line-up of the SFP cleanup system to place cleanup Pump B in a normal line-up, approximately 1800 gallons of water was inappropriately transferred from the SFP to the transfer canal causing level to drop 2 inches. This event was documented in CRDR 2793816.

- April 29, 2005, Unit 2, an actuation of the engineered safety features actuation system (ESFAS) Train B and inadvertent safety injection occurred due to an unintentional de-energization of a cabinet power supply when the redundant power supply was tagged out for maintenance. This event was documented in CRDR 2796508.

- June 2, 2005, Unit 3, while in Mode 3 following a terminated startup, Technical Specification Surveillance Requirement 3.1.2.3, "Verify criticality cannot be achieved with shutdown group CEA movement," as verified by Procedure 72ST-9RX14, "Shutdown Margin - Modes 3, 4, and 5," Revision 9, was not met. This condition occurred when the shutdown group control element assemblies were inserted to less than fully withdrawn with the regulating group control element assemblies not fully inserted. This event was documented in CRDR 2806470.

The inspectors completed five samples.

b. Findings

.1 Use of Refueling Equipment Procedures

Introduction. Three examples of a Green self-revealing NCV were identified for the failure to follow procedures for refueling equipment. The examples involved: (1) a failure to perform a functional test, (2) the failure to cease operation of refueling equipment following the discovery of an equipment deficiency, and (3) the movement of an incorrect spent fuel assembly.

Description. On April 18, 2005, maintenance was performed on the SFHM that affected the up-limit calibration, which impacted the limit switch associated with stopping upward movement of a spent fuel assembly. Specifically, the replacement of the machine coupling following completion of core offload during Refueling Outage 2R12 resulted in misalignment of the up-limit switch. In accordance with the work control process, WO 2781146 was awaiting an SFHM functional retest to verify equipment operability. Fuel services personnel were not aware that the maintenance had been performed and that a retest was required and, consequently, commenced spent fuel movement with a degraded SFHM.

On April 19, the SFHM driver raised the initial spent fuel assembly and slowed the drive motor as the machine approached the point where the up-limit should actuate, as indicated by the Durant counter, stopping further vertical movement and illuminating the up-limit light. The up-limit light did not illuminate as expected, so the driver stopped any further vertical movement by releasing the handswitch. An overload condition also occurred which stopped any further vertical movement. After discussion of the equipment condition with the limited senior reactor operator (LSRO), direction was given to lower then slowly raise the spent fuel assembly to the point where the Durant counter indicated “zero” to confirm that the up-limit was not working correctly. The LSRO acknowledged that the up-limit was not working correctly when it failed to actuate the second time. Consequently, fuel handling activities were stopped and the spent fuel assembly was placed in a safe condition. Procedure 40DP-9OP02, “Conduct of Shift Operations,” Revision 31, Appendix B, requires that the LSRO ensure that the fuel is in a safe condition, stop fuel handling operations, contact the shift manager, and determine the need to complete an event recovery checklist when a deficiency is identified with fuel handling equipment. The procedure further states that the LSRO and shift manager should err on the side of conservatism with the decision to complete the event recovery checklist to ensure that all potential impacts of the identified deficiency on the safety of the fuel are identified and resolved prior to continuing with movement of the affected fuel. Based on these requirements, the LSRO should have ceased operation of the refueling equipment following the initial indication of an SFHM deficiency (up-limit light did not function and an overload condition occurred).

On April 21, the licensee was supporting Westinghouse fuel inspections during Refueling Outage U2R12. A sensitive issues brief was performed prior to the evolution where the importance of reactivity management was stressed. Procedure 78OP-9FX03, “Spent Fuel Handling Machine,” Revision 29, was used to perform the fuel handling operations, which included numerous procedural requirements to ensure that the spent fuel was subject to detailed transfer and inventory control processes. Specifically, Procedure 78OP-9FX03, step 4.1.2 required that, “Anytime fuel is being moved, verify a copy of an independently verified fuel move short form is on the SFHM for the operators’ use. The short form can be obtained from the material balance area (MBA) transfer set package. The operators shall ensure the location obtained from the MBA administrator agrees with the location from the short form.” This requirement was added as a previous corrective action to allow the driver/peer checker to provide an independent check of the LSRO’s selection of SFP location. Additionally, step 4.2.6 required that the operator, “Verify the bridge and trolley are over the specified fuel assembly to be moved using bridge/trolley coordinates.” Procedure 78OP-9FX03, Appendix A, discussed SFHM personnel responsibilities and stated, in part, that “The second person on the SFHM will be the communicator and is responsible for second verification of positioning of the SFHM as it retrieves and discharges assemblies as required; and the philosophy for success is to have two knowledgeable qualified operators on the machine each with distinctive roles, yet working as a peer checking team.”

The MBA transfer form set specified that Fuel Assembly P2P615 be grappled from SFP location Y-25 to support the Westinghouse fuel inspections. The MBA short form was not present on the SFHM as required by Procedure 78OP-9FX03. Instead, the LSRO, SFHM driver, and peer checker collectively reviewed the short list present in the MBA package located near the SFP. The operators returned to the SFHM and verified that the machine was at the correct SFP location. The machine operator then lowered and grappled the fuel assembly, raised the fuel assembly to the up-limit, and moved it to the visual inspection area. The Westinghouse fuel inspector indicated that the number on the assembly did not match the anticipated serial number. Investigation revealed that

Fuel Assembly P2C021 was inappropriately grappled from SFP location Y-26 and moved to the visual inspection stand. Upon recognition that the wrong fuel assembly had been grappled, all fuel handling activities were suspended and the shift manager was immediately notified.

The licensee's investigation concluded that the event resulted from performance deficiencies involving inadequate self, peer, and independent verification practices. Communication of the proper SFP location (Y-25) was overheard by numerous individuals not involved in the fuel movement; however, the incorrect fuel assembly in SFP location Y-26 was ultimately grappled.

Analysis. The performance deficiency associated with the examples of this finding was the failure to follow fuel handling procedures. The finding is greater than minor since it could become a more significant safety concern in that improper handling of spent fuel increases the likelihood of a fuel handling accident. 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 affects the barrier integrity cornerstone and is determined to be of very low safety significance by NRC management review because it was a deficiency that did not result in the actual degradation of spent fuel. This issue involved human performance crosscutting aspects associated with operator decision making, not following procedures, and the use of adequate self, peer, and independent verification practices. This issue also involved problem identification and resolution crosscutting aspects associated with the failure to correct a condition adverse to quality since there have been similar occurrences where operators failed to recognize the need to perform the event recovery checklist (NCV 05000528/2004003-04).

Enforcement. 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. WO 2781146 required the performance of an SFHM retest to verify equipment operability. On April 19, 2005, the licensee commenced the movement of spent fuel without completing the functional test. Procedure 40DP-9OP02, "Conduct of Shift Operations," Revision 31, Appendix B, required that the LSRO ensure that the fuel is in a safe condition, stop fuel handling operations, contact the shift manager, and determine the need to complete an event recovery checklist when a deficiency is identified with fuel handling equipment. On April 19, 2005, fuel handling personnel continued to handle spent fuel even though there was an indication of an SFHM deficiency. Procedure 78OP-9FX03, step 4.1.2, required, anytime fuel is being moved, to verify that a copy of an independently verified fuel move short form is on the SFHM for the operators' use. The short form can be obtained from the MBA transfer set package. The operators shall ensure the location obtained from the MBA administrator agrees with the location from the short form. Additionally, step 4.2.6 required the operator to verify that the bridge and trolley are over the specified fuel assembly to be moved using bridge/trolley coordinates. On April 21, 2005, the MBA short form was not present on the SFHM to perform proper independent verification and operators failed to verify that the bridge and trolley were over the correct fuel assembly. Because the finding is of very low safety significance and has been entered into the CAP as CRDRs 2791974 and 2792326, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2005003-03, “Failure to Implement Procedures for Handling Spent Fuel.”

.2 Inadvertent Reduction of Spent Fuel Water Level

Introduction. A Green self-revealing NCV of Technical Specification 5.4.1.a was identified as a result of the licensee’s failure to follow procedures, which resulted in an inadvertent reduction of SFP water level.

Description. On April 23, 2005, a routine line-up of the SFP cleanup system was in progress to place cleanup pump Train B in a normal line-up per Procedure 40OP-9PC06, “Fuel Pool Cleanup and Transfer,” Revision 34, Appendix AU. Two operators performed the line-up, one manipulating the valves and the other providing independent verification. Once the valve line-up was established, cleanup pump Train B was started. The line-up should have placed the SFP in recirculation.

As a result of corrective actions from past SFP water inventory problems, a third auxiliary operator was stationed at the SFP to locally monitor water level. This operator determined that level was lowering and promptly communicated the problem to the operators performing the line-up. The auxiliary operators performing the line-up responded to the SFP and verified that level was going down. Additionally, a fuel services employee assigned other activities in the area noted that the transfer canal level was increasing and informed the auxiliary operators. The control room was notified and directed that the cleanup pump be secured. The transfer of inventory was stopped when the pump discharge valve was closed. The event resulted in SFP level lowering from 138' 2" to 138', which corresponds to approximately 1800 gallons of water being directed to the transfer canal.

Subsequent investigation by the licensee determined that Valve PCN-V119, “Cleanup Header Return to the Fuel Canal,” and Valve PCN-V080, “Spent Fuel Pool Cleanup Header Return Isolation,” were improperly aligned. As a result of the incorrect alignment, SFP inventory was pumped directly into the transfer canal. Further review by the licensee determined that operations personnel committed several errors. The auxiliary operator performing the line-up did not use appropriate self-checking techniques to reposition the valves. This auxiliary operator had the procedure “in hand” but opened Valve PCN-V119 when the step required the valve to be closed. The auxiliary operator then went to the next step and did not open Valve PCN-V080 as required by the procedure. The auxiliary operator performing the independent verification checked the position of the valves as being correct when in fact they were in the incorrect position. This auxiliary operator also failed to use appropriate self-checking and verification practices.

Analysis. The deficiency associated with the finding was the failure to follow the procedure to line up the SFP cleanup system. The finding is greater than minor because it affects the configuration control 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 procedure implementation and operator attention to detail.

Enforcement. 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 3.h, requires procedures for operating the SFP cleanup system. Procedure 40OP-9PC06, “Fuel Pool Clean Up and Transfer,” Revision 34, Appendix AU establishes the flowpath from Ion Exchanger B to the SFP. Specifically, Appendix AU required: (1) Valve PCN-V119 be closed and Valve PCN-V080 be opened, and (2) that the valves were independently verified in the correct position. Contrary to this, operations personnel did not ensure that Valve PCN-V119 was closed and Valve PCN-V080 was open. Because the finding is of very low safety significance and has been entered into the CAP as CRDR 2793816, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2005003-04, “Failure to Follow Procedures Resulting in Spent Fuel Pool Draindown.”

.3 Engineered Safety Feature Actuation System Train B Actuation

Introduction. A Green self-revealing finding was identified as a result of the licensee’s failure to properly sequence work to maintain power to ESFAS cabinet Train B.

Description. On April 29, 2005, Unit 2 experienced an invalid ESFAS Train B actuation when operations personnel prematurely implemented a tagout permit, causing a loss of vital power to ESFAS cabinet Train B. Loss of power to the cabinet generated the following Train B actuation signals: (1) recirculation actuation signal, (2) auxiliary feedwater actuation signal, (3) containment spray actuation signal, (4) main steam isolation signal, (5) containment isolation actuation signal, (6) safety injection actuation signal, (7) auxiliary feedwater actuation signal, and (8) emergency diesel generator start signal. Since Unit 2 was in the Train B work window for Refueling Outage 2R12, most of the Train B equipment did not actuate because it was tagged out of service for maintenance. The equipment that did actuate were high pressure safety injection Train B and the control room essential ventilation system Train B. All Train B equipment operated as expected. Water from high pressure safety injection Train B injected into the reactor vessel, but the significance was limited since the reactor was defueled with the refueling cavity filled.

Vital power to ESFAS cabinet Train B is maintained by two redundant power supplies, PNC-D27 and PND-D26. The complete loss of vital power was due to Power Supply PNC-D27 being removed from service by Tagging Permit 111894 before Power Supply PND-D26 was restored from maintenance.

Work control personnel recognized the need to restore Power Supply PND-D26 prior to removal of Power Supply PNC-D27, but this need was not communicated to the oncoming shift. The licensee also identified that there was not a technical document or caution to ensure the availability of the redundant power supply to the ESFAS cabinet. The sequencing of the permits resulted in the de-energization of both power supplies and the ESFAS actuation.

Analysis. The performance deficiency associated with this finding was a failure to control work during maintenance evolutions. The finding is greater than minor since it was associated with the configuration control attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the reliability and availability of systems that respond to initiating events. This finding cannot be evaluated by the significance determination process because Manual Chapter 0609, "Significance Determination Process," Appendix G, "Shutdown Operations Significance Determination Process," do not apply when the reactor is defueled. This finding is determined to be of very low safety significance by NRC management review because it was a deficiency that did not result in actual safety consequences since the reactor was defueled and a majority of the Train B equipment was tagged out for maintenance. This issue involved human performance crosscutting aspects associated with inadequate communications between work control groups and awareness of plant configuration.

Enforcement. No violations of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because the tagout permit was appropriate. Rather, operations personnel prematurely implemented a tagout permit, causing loss of vital power to ESFAS cabinet Train B. This finding has been entered into the licensee’s CAP as CRDR 2796508, Finding (FIN) 05000529/2005003-05, “Inadvertent ESFAS Actuation.”

## 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 Updated Final Safety Analysis Report 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 Technical Specifications; (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.

- April 2, 2005, Unit 2, failure of main steam line hanger described in CRDR 2783278
- April 19, 2005, Units 1, 2, and 3, cover plate bolt failures on recirculation sump suction isolation valve described in CRDRs 2791717, 2792201, and 2792201

The inspectors completed two samples.

b. Findings

Introduction. A Green self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified for the failure to implement corrective actions to preclude repetition of a significant condition adverse to quality. Specifically, the licensee’s corrective actions for replacing susceptible fasteners on 16-inch valves did not address the need to replace similar susceptible fasteners on 10-inch and 24-inch valves.

Description. On March 19, 1988, a leak was discovered in a shutdown cooling system heat exchanger outlet valve. Investigation revealed that the bolting which secures a gasket retainer plate had fractured. The engineering root cause evaluation determined that the bolts did not meet specification requirements and that they had failed as a result of stress corrosion cracking. Similar bolt fractures occurred in Unit 3 on March 21, 1987, and in Unit 2 on June 2, 1988. This event was reported in Licensee Event Report (LER) 50-528/1988-022-00. These failures occurred on 16-inch butterfly valves with half-inch gasket retainer bolts. Corrective actions to prevent recurrence following this event were to replace the susceptible bolting material for the 16-inch butterfly valves with half-inch gasket retainer bolts. However, an inadequate transportability review resulted in the failure to consider other valves of similar design from the same vendor.

On April 19, 2005, during a planned inspection of the recirculation sump, the licensee found three of the four gasket retainer plate bolts sheared on Valves 2JSIAUV673 and 2JSIBUV675, which are the inboard 24-inch butterfly containment sump suction isolation valves. The bolts were 3/8-inch stainless steel bolts that secured the gasket retainer plate to the valve body. Analysis by the licensee determined that the bolts failed due to stress corrosion cracking. Stress corrosion cracking occurs if there is a susceptible material under stress and a wetted surface. The licensee determined that these bolts had the incorrect surface tempering, since the measured hardness was too high. Additionally, these bolts had been submerged in borated water as a result of corrective actions implemented following identification of the voiding concern with the sump suction piping as described in NRC Inspection Report 05000528; 05000529;

05000530/2004014. The licensee also determined that the 10-inch containment spray to shut down cooling heat exchanger valves were susceptible. Corrective actions following identification of the bolt failures were to replace all bolts with the correct material.

Analysis. The performance deficiency associated with this finding was the failure to take adequate corrective actions to preclude repetition of a significant condition adverse to quality. The finding is greater than minor since it affected the equipment performance attribute of the mitigating systems 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 the failure to perform an adequate transportability review.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, 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. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, in 1988 the licensee identified that the gasket retaining bolts on several butterfly valves were susceptible to stress corrosion cracking. The licensee only replaced bolts on the 16-inch valves with the identified failures and did not consider the need to replace bolts on similarly designed 10-inch and 24-inch valves. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2791716, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2005003-06, "Failure to Take Adequate Corrective Actions to Prevent Bolt Failures."

## 1R16 Operator Workarounds (71111.16)

### .1 Review of Operator Workarounds

#### 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 had identified and implemented appropriate corrective actions associated with operator workarounds.

- May 17, 2005, Unit 2, broken handswitch knob on Breaker NAN-S05D

The inspectors completed one sample.

b. Findings

Introduction. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the failure to follow the procedure to properly track an operator workaround and implement appropriate compensatory measures for a broken handswitch.

Description. On March 17, 2005, during a walkdown of the Unit 2 control room, the inspectors identified that the knob associated with the handswitch for the normal supply from offsite power Breaker NAN-S05D was broken and was not tracked as an operator workaround.

Administrative Procedure 40DP-9OP15, "Operator Challenges and Discrepancy Tracking," Revision 15, defines an operator workaround as an operator challenge that affects transient plant operations requiring operators to perform compensatory actions in order to comply with an emergency operating or abnormal operating procedure. Procedure 40DP-9OP15 specifies that an operator workaround is a subcategory of an operator challenge.

Emergency Operating Procedure 40EP-9EO10, "Standard Appendices," Revision 35, Appendix 67, requires operation of Breaker NAN-S05D to restore offsite power following a loss of offsite power event. The inspectors questioned the operators regarding the method that would be used to operate the handswitch. The operators replied that they could operate the handswitch by removing the broken knob and turning a metal tab on the switch. They stated that the metal tab could be turned using a pair of pliers or by a spare knob located somewhere in the control room to compensate for the broken knob. Operators were not able to immediately locate either of these tools in the control room. Operators eventually located the spare knob. Knobs in the control room associated with breaker handswitches have broken in the past, but were not tracked as operator workarounds. The inspectors determined through questioning several operators that there was no consistent guidance regarding the use of pliers or a spare knob to operate a breaker handswitch when the knob was broken.

Analysis. The performance deficiency associated with this finding was the failure to properly track and control tools necessary to operate breaker handswitches when the knob was broken. The finding is determined to be greater than minor because if left uncorrected it could become a more significant safety concern in that operators may not be able to operate equipment necessary to respond to initiating events. The finding affects the equipment performance attribute of the mitigating systems cornerstone objective to ensure the reliability and availability of systems that respond to initiating events. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," the finding is determined to have very low safety significance because it only affects the mitigating systems cornerstone and did not result in the actual loss of a safety function.

Enforcement. 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Administrative Procedure 40DP-9OP15, "Operator Challenges and Discrepancy Tracking," Revision 15, Section 2.1.1, stated in part that "An Operator Challenge is an equipment or program deficiency that provides an obstacle to safe plant operations requiring operators to perform compensatory actions." Section 2.1.3 stated that "All plant or equipment deficiencies that meet the definition of an Operator Challenge shall be identified and tracked to resolution." Contrary to these requirements, the licensee failed to identify and track an operator challenge. Specifically, an operator workaround to control tools necessary to compensate for a breaker handswitch with a broken operating knob was not identified and tracked. Because this finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2807501, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2005003-07, "Failure to Identify an Operator Challenge for a Broken Switch."

.2 Cumulative Review of the Effects of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of operator workarounds to determine: (1) the reliability, availability, and potential for misoperation of a system; (2) if multiple mitigating systems could be affected; (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents; and (4) if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

Annual Review

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flowpaths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the modification listed below. The inspectors verified that: (1) modification preparation, staging, and implementation does not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; (2) postmodification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, SSC performance characteristics still meet the design basis, modification design assumptions are appropriate, and the modification test acceptance criteria has been met; and (3) the licensee has identified and implemented appropriate corrective actions associated with permanent plant modifications.

- April 15, 2005, Unit 3, Design Modification WO 2473654, “Add a Redundant 125VDC/24VDC Power Supply to Diesel Generators A and B Governor Control Circuits”

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the four below listed postmaintenance 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.

- May 4, 2005, Unit 1, retest of the balance of plant ESFAS Train A sequencer per Work Mechanism (WM) 2796881

- May 6, 2005, Unit 2, retest following repair of emergency diesel generator overspeed butterfly valve per WM 2797828

- May 25, 2005, Unit 2, retest following repair of Valve SGHV-0500R per WM 2777042

- May 27, 2005, Unit 3, retest following repair of Valve 3PSIE-217 per WOs 2802157 and 2802158

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Refueling Outage 2R12

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan and compliance with the Technical Specifications: (1) the risk control plan, (2) tagging/clearance activities, (3) reactor coolant system (RCS) instrumentation, (4) electrical power, (5) decay heat removal, (6) spent fuel pool cooling, (6) inventory control, (7) reactivity control, (8) containment closure, (9) reduced inventory or midloop conditions, (10) refueling activities, (11) heatup and cooldown activities, and (12) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Unit 3 Short Notice Outage

For the below listed outage, 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 Technical Specifications: (1) the risk control plan, (2) electrical power, (3) decay heat removal, (4) reactivity control, (5) containment closure, (6) heatup and cooldown activities, and (7) licensee identification and implementation of appropriate corrective actions associated with outage activities.

- May 23 through June 25, 2005, Unit 3 short notice outage to replace damaged pressurizer heaters and repair oil leaks on two reactor coolant pump thrust bearings

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 Updated Final Safety Analysis Report, procedure requirements, and Technical Specifications to ensure that the four below listed surveillance activities demonstrated that the SSCs 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 Technical Specification 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.

- April 3, 2005, Unit 2, Procedure 73ST-9DG01, “Class 1E Diesel Generator and Integrated Safeguards Test, Train A,” Revision 9, Section 8.6

- April 21-22, 2005, Unit 2, local leak rate testing of Containment Penetrations 38 and 52 per Procedure 73ST-9CL01, “Containment Leakage Type ‘B’ and ‘C’ Testing,” Revision 25, Sections 8.16 and 8.28

- May 29, 2005, Unit 3, Procedure 32ST-9RC01, “92 Day Pressurizer Heater Functional Test,” Revision 9

- June 15, 2005, Unit 1, Procedure 73ST-9AF03, "AFB-P01 Inservice Test," Revision 15, Section 8.1, and Procedure 73ST-9XI38, "AF Pumps Discharge Check Valves - Inservice Test," Revision 12, Section 8.2

The inspectors completed four samples.

b. Findings

Introduction. A Green self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified for an inadequate surveillance procedure which resulted in an inadvertent safety injection and subsequent RCS level transient.

Description. On April 3, 2005, operations personnel performed Procedure 73ST-9DG01, “Class 1E Diesel Generator and Integrated Safeguards Test, Train A,” Revision 9. Procedure 73ST-9DG01, step 8.6.27, simulates a loss of power combined with a safety injection actuation signal and containment isolation actuation signal. Upon activation of the Train A components, a pressurizer level increase was noted.

Immediate actions taken by operations personnel were to secure the Train A safety injection pumps, which was a contingency action discussed at the prejob briefing.

Review of the level trends indicated that the source of the water ingress was from safety injection Tanks (SITs) 1A and 1B. Both SITs 1A and 1B were at ambient pressure (14.7 psia), the vent valves were open to containment atmosphere, and level was approximately 80 percent. RCS pressure was approximately 18.7 psia. When the SIT outlet isolation valves opened, SIT level decreased approximately 6 percent and pressurizer level increased approximately 14 percent (approximately 2000 gallons of water transferred). The water transfer occurred since the total pressure head in the SITs (approximately 20.3 psia) was higher than the RCS. Once the SIT outlet isolation valves were reclosed, RCS water inventory stabilized.

Procedure 73ST-9DG01 cautioned operations personnel to evaluate the pressure difference between the RCS and SITs prior to any actuation that opened the SIT outlet isolation valves. The procedure was inadequate in that it failed to have operations personnel consider level differences which could potentially impact the total pressure head of the system.

Analysis. The performance deficiency associated with this finding was an inadequate procedure. The finding is determined to be greater than minor because it affected the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Using Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," this finding is determined to have very low safety significance because the event did not constitute a loss of control and did not represent a finding requiring quantitative assessment. The finding did not increase the likelihood of loss or cause a degradation in the ability to restore decay heat removal, RCS inventory, offsite power, alternate core cooling, or containment. This issue involved human performance crosscutting aspects associated with inadequate operations procedures.

Enforcement. 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, Procedure 73ST-9DG01, "Class 1E Diesel Generator and Integrated Safeguards Test," Train A, Revision 9, did not contain the appropriate steps or cautions to preclude an inadvertent safety injection. Because this finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 2786378, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2005003-08, "Inadvertent Safety Injection During Integrated Safeguards Testing."

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, plant drawings, procedure requirements, and Technical Specifications 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 postinstallation test results were satisfactory and that the impact of the temporary modification on permanently installed SSCs 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.

- May 28, 2005, Unit 3, Temporary Modification 2802998, “Installation of Oil Collection Funnels on RCP [reactor coolant pump] Lube Oil Collection Piping”

The inspectors completed one sample.

b. Findings

No findings of significance was identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector performed an in-office review of Revisions 30 and 31 to the Palo Verde Nuclear Generating Station Emergency Plan, submitted April 11and March 4, 2005. The inspector also performed an in-office review of Revisions 3 and 4 to Emergency Plan Implementing Procedure 99, Appendices A and P. These revisions:

- Added additional descriptions of the normal station organization, the duties of several emergency response organization positions, the protective action recommendation process, of the emergency response facilities, and the training program for emergency response organization personnel

- Removed one position from the Emergency News Center

- Revised Emergency Action Levels 3-16 and 3-19 from "Field survey result or valid dose assessment indicates > 100/1000 mrem TEDE [total effective dose equivalent] or > 500/5000 thyroid CDE [committed dose equivalent] at the Site

Boundary," to “Site Boundary dose rate > 100/1000 mrem/h DDE [deep dose equivalent] as measured with portable instrumentation OR valid dose projection > 100/1000 mrem TEDE or > 500/5000 thyroid CDE at the Site Boundary.”

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, Nuclear Energy Institute 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.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Reviews

a. Inspection Scope

In order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s CAP. This review was accomplished by reviewing daily CRDR summary reports and attending CRDR review meetings. 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.

b. Findings and Observations

On May 17, 2005, the licensee identified that the mechanical seal for the Unit 2 low pressure safety injection (LPSI) pump Train A had been leaking. The unit was in Mode 3 when the condition was discovered. The licensee declared the pump inoperable upon discovery, entered Technical Specification Limiting Condition for Operation 3.5.3, Condition A, and initiated Work Mechanism 2800334 to replace the seal.

The inspectors reviewed this equipment failure during a plant status tour on May 18, 2005, and noted that the LPSI Train A pump was secured while Unit 2 was in Mode 4 on May 15 and placed in standby per the normal operating lineup. It appeared that the leakage identified by the licensee had occurred while the pump was operating, which indicated that the pump may have been inoperable prior to entering Mode 3 on May 16, 2005. The inspectors also noted that the licensee was focused on equipment repairs and had not recognized the need to initiate a CRDR to consider the potential reportable condition and equipment cause of failure investigation. Following the inspectors’ identification of this oversight to the licensee, CRDR 2800779 was written to perform the reportability review and maintenance rule functional failure determination. On June, 10, 2005, the licensee determined that the mechanical seal failure constituted a maintenance rule functional failure and was caused by inadequate venting of the seal upon restoration to service in Mode 5. Furthermore, the licensee determined that LPSI pump Train A was not operable when applicable Technical Specification Limiting Condition for Operation 3.5.3 Mode 3 conditions were entered on May 16, 2005.

.2 Assessment of Licensee Trending Programs

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspector performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspector's review was focused on human performance and configuration management, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspector's review nominally considered the 7-month period of October 2004 through April 2005, although some examples expanded beyond those dates when the scope of the trend warranted. The inspector reviewed approximately 50 specific CAP documents, including those associated with human performance and configuration management events that occurred during the period and those written to document station trends. The review also included issues documented outside the normal CAP in system health reports, self-assessment reports, and various trend reports published by the Performance Improvement Department (PID) and the Communications Department. The specific items reviewed are listed in the Documents Reviewed section attached to this report. Corrective actions associated with the issues identified in the licensee's trend report were reviewed for adequacy.

b. Findings and Observations

The inspector evaluated the licensee's CAP trending methodology and observed that the licensee had performed a detailed review, but weaknesses were noted in the program as described below. The licensee routinely assigns cause codes and responsible organizations, but does not always use this information to identify potential trends in their CAP data. Trend analysis is based upon categories into which individual corrective action documents are placed. The number of occurrences in each category is then compared to an upper control limit (UCL) as one indication of whether or not an adverse trend of performance exists in that category. The inspector noted the following weaknesses in this approach:

(1) Institutionalized trending does not normally occur based on human performance cause codes alone. Cause codes are more frequently used to understand site-wide issues that are identified elsewhere. For example, the licensee performed a site-wide evaluation of occurrences of Cause Code WP08, "Documents Not Followed Correctly," only as a result of a Common Cause

Investigation Report (CRDR 2780273), which was conducted to address the human performance crosscutting issue identified in the NRC's 2004 Annual Assessment Letter. In this review, the licensee reported that a 2003 CRDR identified that ". . . the approach to resolving [WP08] has been fragmented, with little integration of the efforts on a cross-organization basis." The review went on to state that at present WP08 ". . . continued to be dealt with on an organization-by-organization basis rather than as a site-wide issue." The process of evaluating trends only by categories and not by cause codes is a programmatic weakness that may mask trends that exist across organizational lines. This conclusion is supported by data that was provided to the inspector. During the one year period of April 1, 2004, through March 31, 2005, Cause Code WP08 was applied to 224 CRDRs, making WP08 the second most common cause code applied. Expanding trending efforts to consider cause codes may have provided an earlier indicator of this weakness which was eventually identified in CRDR 2780273.

(2) The effectiveness of the licensee's CAP trending program is subject to the quality of the UCL. The original UCLs assigned to each category were derived from historical norms, but do not appear to have been recalculated to make them useful indicators of developing trends. As an example, the UCL for the configuration management category was reported as 111 occurrences per month in the March 2005 Main Category - Upper Control Limit Report. In the preceding 6 months, the average number of configuration management related CRDRs was 33.8 per month with a high of 51. As a result of the high UCL, reviews conducted in this category have been insensitive to existing or rising trends. This conclusion is supported by an internal audit completed in April 2005 (Audit Report 05-005, "Design Control"), which identified that "trend monitoring did not provide management the visibility needed to see the accumulation of unincorporated and reserved engineering documents." The licensee acknowledged that a periodic review of UCLs had always been intended but not performed in practice. The licensee has initiated CRDR 2806832 to document this weakness and track corrective actions.

(3) The manner in which the licensee defines human performance events could potentially mask trends. In the site-wide Monthly Trend Report, the licensee tracks only "significant" human performance events (HPEs). In addition, the most recent PID Trend Report referred to the number of significant and "noteworthy" HPEs in recent months to justify the conclusion that operations human performance was improving and that corrective actions had been effective. This conclusion was reached despite the fact that the Operations Department human performance annunciator window went from "green" to "yellow" in the last month of the report period. The reliance on the defined HPE categories could provide a potentially misleading view of actual human performance, as born out by events in the April 2005 refueling outage. A sample of approximately 20 CRDRs reviewed by the inspector from the April 2005 refueling outage contained nine human performance errors, including a failure to recognize alarming reactor coolant pump oil levels, inadvertent draining of the

RCS during refill, and valves left out of position that caused an inadvertent draindown of approximately 1800 gallons from the SFP. Due to the fact that none of these nine HPEs were classified as "significant," none of these errors contributed to the site-wide April 2005 Monthly Trend Report and will not contribute to the data historically considered in the PID Trend Report for the Operations Human Performance trend.

(4) PID trending efforts do not appear to be making an accurate assessment of site-wide performance in Permit and Tagging. An adverse trend in the area of permit and tagging errors has been carried in the PID Trend Report dating back to 2003. As documented in CRDR 2742173, this trend was elevated to the Nuclear Assurance Department's site-wide Top Ten list in 2004, which resulted in a number of initiatives to improve performance. One of these initiatives was the establishment of an Operations Department annunciator window for permit and tagging, which was reported as red in 2 of the 3 months of the fourth quarter of 2004. The licensee attributed these red performance results to an "outage spike" in November 2004 and to poor performance by the Water Reclamation Facility and Maintenance Department for December 2004. Despite these results, PID's Trend Report for the fourth quarter of 2004 concluded that station performance in permit and tagging was improving. In fact, three departments have been reported as "red" in the new tagging and permit annunciator window in recent months (Operations, Water Reclamation Facility, and Maintenance (twice)), suggesting that the site-wide performance may not actually be improving as stated in PID's Trend Report.

In addition, in the licensee's Monthly Trend Reports produced by the Communications Department, the facility is reported as having Green performance in tagging and permit events for the months of January, February, March, and April 2005. In data provided to the inspector, the licensee reported 31 occasions between October 2004 and March 2005 where CRDR Cause Code WP02, "System Alignment, Tagout and Restoration," was assigned to CRDRs, compared with a total of nine adverse or significant tagging events as reported in the Monthly Trend Reports covering the same period. In a population of approximately 20 configuration management CRDRs reviewed by the inspector during the April 2005 refueling outage, four involved aspects of the tagging and clearance process, none of which contributed to the April Monthly Trend Report.

These examples suggest that the tagging and permit trend may be adverse despite the conclusions drawn by the Monthly Trend Report or the PID Quarterly Trend Report.

The inspectors noted that, of the four occasions in which departments were reported as red on the tagging and permit annunciator window, three CRDRs were written as required by the Top Ten Action Plan. Two of these CRDRs are now closed with all actions complete (Maintenance Department and Water Reclamation Facility). One CRDR (Operations Department) was initiated in

June 2005 based on red performance in May. The fourth red occurrence was a repeat occurrence for the Maintenance Department in April 2005. The inspectors noted that none of the three CRDRs written for tagging issues suggested any site-wide actions or reviews. Lastly, the inspectors noted that CRDR 2742173, written to document the actions taken for the Top Ten Action Plan, was closed in April 2005 with no remaining action items. An effectiveness review conducted under CRDR 2742173 stated that corrective actions had been effective and that the Nuclear Assurance Department intends to remove permit and tagging from the site-wide Top Ten Action Plan.

The inspectors compared the licensee process results with the results of the inspectors' daily screening and the results of the more detailed review and did not identify any potential trends in the CAP data that the licensee had failed to identify.

.3 Assessment of Licensee Actions for Main Steam Safety Valve (MSSV) Failures

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspector performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspector's review was focused on MSSV failures and included corrective action documents and LERs written between 1999 and April 2005 and a review of NRC generic communications in the area of MSSV performance. The review also included issues documented outside the normal CAP in system health reports, self-assessment reports, and various trend reports provided to management. The specific items reviewed are listed in the Documents Reviewed section attached to this report.

b. Assessments and Observations

Prior to each outage, PVNGS performs lift testing of each main steam safety valve, during which each valve must lift within plus or minus 3 percent of its nominal setpoint. The number of valves that failed to meet these criteria has declined over recent years due to industry-wide and site-specific actions.

After a relatively large number of failures in 1999, the licensee performed a significant equipment root cause of failure analysis investigation to determine causes and mitigation strategies for MSSV sticking. In this evaluation, the licensee provided a thorough discussion of the types of failures experienced, the possible causes, industry operating experience, and suggested corrective actions. The licensee separated the MSSV failures into two categories: sticking and setpoint drift.

An exhaustive explanation of the conditions that contribute to MSSV sticking included phenomena not isolated to Palo Verde (for example, microbonding between disc and seat, moisture carryover, and iron concentration in the secondary systems). The licensee participated in EPRI research on the issue and visited other nuclear facilities to benchmark their performance. Corrective actions for valve sticking appear to have been appropriate, as the number of "sticking" failures that have occurred since 1999 has dropped significantly.

The licensee provided two possible causes for setpoint drift: test equipment ramp speed and thermal effects on valve components. Both were explained in some detail in the investigation report; however, the licensee concluded that neither of these was of sufficient magnitude to cause drift failures to occur. As a result, the licensee elected in 1999 to take no corrective action to prevent repetition of drift failures.

Despite the licensee's conclusion that the causes identified could not lead to the drift failure mechanism, additional drift failures have occurred. LER 529-2002-001 reported the failure of two MSSVs by the drift mechanism. A significant root cause investigation report was completed with the charter of determining root causes and corrective actions to prevent repetition. This report did not arrive at a discrete cause for the event and did not suggest any corrective actions to preclude repetition.

In the absence of a definitive cause, the licensee assumed that the mechanism behind the setpoint drift failures might be similar to the known causes of MSSV sticking. The licensee replaced both of the MSSVs that had failed during the subsequent outage. The replacement valves included improved seat materials that were designed to minimize thermal effects on the valve seat and disc. Both valves met their acceptance criteria during testing in March 2005.

c. Findings

No findings of significance were identified.

.4 Crosscutting Issues Followup Inspections

The inspectors reviewed CRDRs 2780273 and 2780286, which document the NRC's identification of substantive crosscutting issues in the human performance and problem identification and resolution areas, respectively. The substantive crosscutting issues were documented in NRC Inspection Report 05000528; 05000529; 05000530/2005001, "Annual Assessment Letter - Palo Verde Nuclear Generating Station." The inspectors observed that the common cause evaluations appeared to be narrowly focused. Subsequently, the licensee expanded their analysis and action plan development to correct the identified issues. As of the end of the inspection period, the licensee's re-analyses into the substantive crosscutting issues had not been completed.

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

Sections 1R04 and 1R14.1 describe findings that involved the failure to correct conditions adverse to quality.

Section 1R15 describes a finding that involved an inadequate transportability review.

4OA4 Crosscutting Aspects of Findings

Section 1R04 describes a finding were the licensee failed to identify and correct a condition adverse to quality.

Section 1R14.1 describes a finding where: (1) operator decision making was nonconservative; (2) procedures were not followed; and (3) inadequate self, peer, and independent verification practices were utilized.

Section 1R14.2 describes a finding where inadequate procedure implementation and poor operator attention to detail resulted in an inadvertent loss of SFP inventory.

Section 1R14.3 describes a finding where inadequate communications between work control groups and awareness of plant configuration resulted in an inadvertent ESFAS actuation.

Section 1R22 describes a finding where an inadvertent safety injection actuation signal occurred due to inadequate operations procedures.

4OA5 Other Activities

.1 (Closed) Temporary Instruction 2515/150: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

The first occurrence of TI 2515/150 was documented for Unit 2 in NRC Inspection Report 05000529/2003-005. The second occurrence of TI 2515/150 for Unit 2 is documented below. Therefore, TI 2515/150 is now closed for Units 1, 2, and 3.

a. Inspection Scope

The inspectors observed and reviewed licensee activities associated with the reactor pressure vessel head and vessel head penetration nozzle inspection that were implemented in accordance with the requirements of Order EA-03-009.

The licensee performed ultrasonic and eddy current examinations of all control element drive mechanism penetrations. The inspectors independently reviewed the inspection results. The licensee did not identify any nozzle or weld degradation.

The licensee performed a 100 percent visual inspection of the reactor vessel head. The inspectors reviewed detailed video tapes of the head examination. No flaws were identified.

b. Findings

No findings of significance were identified.

.2 TI 2515/160: Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Reactors

a. Inspection Scope

Implementation of this TI was required for facilities that include Alloy 600 base metal materials or Alloy 82/182 weld metal materials in the design of their pressurizer penetration nozzles, heater sleeves, or steam space piping components. The inspectors reviewed the licensee’s pressurizer design and modification documents and determined that, although the initial design utilized these materials, the licensee had replaced them with Alloy 690. The inspectors reviewed the documents associated with this modification.

b. Findings

No findings of significance were identified.

.3 TI 2515/161: Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

This area was inspected to verify that the licensee’s radioactive material transportation program complies with specific requirements of 10 CFR Parts 20 and 71 and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel and determined the licensee had undergone refueling/defueling activities between January 1, 2002, and present, but it had not shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

.4 TI 2515/163: Operational Readiness of Offsite Power

The inspectors collected data pursuant to TI 2515/163, "Operational Readiness of Offsite Power." The inspectors reviewed the licensee's procedures related to General Design Criteria 17, "Electric Power Systems"; 10 CFR 50.63, "Loss of All Alternating Current Power"; 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"; and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. Documents reviewed for this TI are listed in the attachment.

NRC Augmented Inspection Team Report 05000528; 05000529; 05000530/2004-012 identified several unresolved items following the loss of offsite power event on June 14, 2004. The team identified Unresolved Item 05000528; 05000529; 05000530/2004012-06 to review the grid reliability, independence, and stability for Palo Verde

Nuclear Generating Station. The NRC will perform this review upon issuance of the Western Electric Coordinating Council (WECC) Detailed System Disturbance Report, conducted by the WECC Disturbance Task Force for the Westwing Outage that occurred on June 14, 2004. This unresolved item remains open.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On April 7 and May 2, 2005, the emergency preparedness inspector conducted a telephonic exit meeting to present the inspection results to Mr. E. O'Neill, Department Leader, Emergency Planning, and other members of his staff, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On April 14, 2005, the engineering inspectors presented the results of the inservice inspection effort to Mr. D. Mauldin, Vice President, Engineering and Support, and other members of licensee management. Licensee management acknowledged the inspection findings.

On June 14, 2005, the radiation protection inspector discussed the inspection findings with Mr. J. Gaffney, Manager, Radiation Protection. The inspector verified that no proprietary information was provided during the inspection.

On June 28, 2005, the resident inspectors presented the inspection results of the resident inspections to Mr. D. Mauldin, Vice President, Engineering and Support, and other members of the licensee management. The licensee acknowledged the findings presented.

The inspectors noted that while proprietary information was reviewed, none would be included in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

S. Bauer, Department Leader, Regulatory Affairs
P. Borchert, Director, Work Management
R. Buzard, Sr. Consultant, Regulatory Affairs
D. Carnes, Director, Nuclear Assurance
P. Carpenter, Unit Department Leader, Operations
C. Churchman, Director, Engineering
S. Coppock, Department Leader, System Engineering
E. Dutton, Section Leader, Performance Improvement
D. Fan, Department Leader, Design Mechanical Engineering
J. Gaffney, Director, Radiation Protection
T. Gray, Radiological Services Department Leader, 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
G. Overbeck, Senior Vice President, Nuclear Operations
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

Other Personnel

E. Shouse, Representative, El Paso Electric Representative
J. Taylor, Nuclear Project Manager, Public Service of New Mexico

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000528; 05000529; 05000530/2005003-01 | NCV | Failure to Correct a Condition Adverse to Quality (Section 1R04) |
| 05000528/2005003-02 | NCV | Maintenance Performed on Fuel Handling Equipment Without Proper Qualifications (Section 1R12) |
| 05000529/2005003-03 | NCV | Failure to Implement Procedures for Handling Spent Fuel (Section 1R14.1) |
| 05000528/2005003-04 | NCV | Failure to Follow Procedures Resulting in Spent Fuel Pool Draindown (Section 1R14.2) |
| 05000529/2005003-05 | FIN | Inadvertent ESFAS Actuation (Section 1R14.3) |
| 05000529/2005003-06 | NCV | Failure to Take Adequate Corrective Actions to Prevent Bolt Failures (Section 1R15) |
| 05000529/2005003-07 | NCV | Failure to Identify an Operator Challenge for a Broken Switch (Section 1R16.1) |
| 05000529/2005003-08 | NCV | Inadvertent Safety Injection During Integrated Safeguards Testing (Section 1R22) |

Closed

None

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

CRDR
2808731

Procedure
40AO-9ZZ21, “Acts of Nature,” Revision 21

Section 1R04:  Equipment Alignment

Procedures
40OP-9PC01, “Fuel Pool Cooling,” Revision 5
40OP-9EW01, “Essential Cooling Water System (EW) Train A,” Revision 6
40OP-9EW02, “Essential Cooling Water System (EW) Train B,” Revision 4
40OP-9SI01, “Shutdown Cooling Initiation,”Revision 33
73ST-9XI38, “AF pumps discharge check valves - inservice test,” Revision 12

Drawings
02-M-PCP-001, “Fuel Pool Cooling & Cleanup System,” Revision 22
02-M-EWP-001, “Essential Cooling Water System,” Revision 25
01-M-NCP-002, “Nuclear Cooling Water System,” Revision 10
02-M-DGP-001, “P&I Diagram Diesel Generator System,” sheet 1 of 9, Revision 42
02-M-DGP-001, “P&I Diagram Jacket Water Diesel Generator System,” sheet 4, Revision 42
02-M-DGP-001, “P&I Diagram Cooling Water Diesel Generator System,” sheet 5, Revision 42
02-M-DGP-001, “P&I Diagram Starting Air Diesel Generator System,” sheet 6, Revision 42
02-M-DGP-001, “P&I Diagram Diesel Generator System,” sheet 9, Revision 42
03-M-SIP-001, “Safety Injection and Shutdown Cooling System,” Revision 25
03-M-SIP-002, “Safety Injection and Shutdown Cooling System,” Revision 22
01-M-AFP-001, “P&I Diagram Auxiliary Feedwater System,” Revision 32
01-M-CTP-001, “P&I Diagram Condensate Storage and Transfer System,” Revision 20

Miscellaneous
Calculation 13-JC-DG-201, “Diesel Fuel Oil Day Tank Level Instrument (DGN-L-1, DGN-L-2, DGN-L-5 & 6) Uncertainty Calculation”

System Training Manual, Volume 21, “Auxiliary Feedwater System (AF)”

Design Basis Manual, Auxiliary Feedwater System (AF), Revision 14

PVNGS Updated FSAR, Section 10.4.9, “Auxiliary Feedwater System”

CRDRs
2800972, 2624895, 9-8-1508, 2721947

Section 1R05:  Fire Protection

Miscellaneous
Palo Verde Nuclear Generating Station prefire strategy
Palo Verde Nuclear Generating Station History Report:  Unit 1 Extinguishers

Palo Verde Nuclear Generating Station History Report: Unit 2 Extinguishers
Palo Verde Nuclear Generating Station History Report: Unit 3 Extinguishers
Material Safety Data Sheet, Chevron GST Oils, Revision 6

CRDRs
2795950, 2798236

Section 1R07: Heat Sink Performance

Work Order
2733162

Miscellaneous
Trend of Spray Pump Flow and Cooling Water flows for April 2, 2005

Section 1R08: Inservice Inspection Activities

Procedures
70TI-9ZC01, "Boric Acid Corrosion Prevention Program," Revision 5
73DP-0EE16, "Qualification and Certification of NDE Personnel," Revision 5
73DP-9XI03, "ASME Section XI Inservice Inspection," Revision 5
73TI-9RC01, "Steam Generator Eddy Current Examinations," Revision 23
73TI-9ZZ32, "Steam Generator Secondary Pressurization Test," Revision 6
73TI-9ZZ78, "Visual Examination for Leakage," Revision 5
81DP-9RC01, "PVNGS Steam Generator Degradation Management Program," Revision 3

Miscellaneous
Letter dated March 25, 2005, Palo Verde Nuclear Generating Station, Unit 2, Docket STN 50-529, 10 CFR 50.55a(3)(I), "Alternative Repair Request for Reactor Coolant System Hot Leg Alloy 600 Small-Bore Nozzles (Relief Request 31)"

"Replacement Steam Generator Preservice Examination Report," dated March 12, 2004

Liquid Penetrant Examinations
PT-05-022
PT-05-023
PT-05-026

Magnetic Particle Examinations
MT-05-008
MT-05-009
MT-05-010
MT-05-011

Radiographic Examinations
219436-7CI

Ultrasonic Examinations

| | | | |
|---|---|---|---|
| UT-03-079-1 | UT-05-026-3 | UT-05-050-1 | UT-05-058-3 |
| UT-03-079-2 | UT-05-026-4 | UT-05-050-2 | UT-05-059-1 |
| UT-03-079-3 | UT-05-048-1 | UT-05-051-1 | UT-05-059-2 |
| UT-03-079-4 | UT-05-048-2 | UT-05-051-2 | UT-05-059-3 |
| UT-05-026-1 | UT-05-049-1 | UT-05-058-1 | UT-05-061-1 |
| UT-05-026-2 | UT-05-049-2 | UT-05-058-2 | UT-05-061-2 |

CRDRs

| | | | |
|---|---|---|---|
| 2711440 | 2744493 | 2746765 | 2776133 |
| 2716347 | 2745744 | 2748924 | 2777240 |
| 2717668 | 2746251 | 2771873 | 2783599 |
| 2722696 | | | |

Section 1R12: Maintenance Effectiveness

Procedures
15DP-0TR69, "Training and Qualification Administration," Revision 10
30DP-9MP01, "Conduct of Maintenance," Revision 37
Palo Verde Updated Final Safety Analysis Report, Chapter 17

CRDRs
113786, 2797536

Miscellaneous
APS Letter 102-05267-GRO/CKS/REB, dated May 6, 2005, "Palo Verde Nuclear Generating Station (PVNGS) Response to Allegation RIV-2005-A-0030"

Work Orders

| | | |
|---|---|---|
| 2692669 | 2781144 | 2792044 |
| 2744188 | 2788705 | |
| 2774202 | | |

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
72ST-9RX03, "DNBR/LHR/AZTILT/ASI with COLSS out of service," Revision 10

Work Order
2800334

Miscellaneous
10 CFR 50.59 screening S-05-0067
10 CFR 50.59 evaluation E-05-0010

CRDRs
2790940, 2791164, 2789622

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

Procedures
40OP-9PC06, “Fuel Pool Cleanup and Transfer,” Revision 34

CRDRs
2793816, 2807195, 2790326, 2798080, 2791515

Section 1R16: Operator Workarounds

Procedures
40DP-9OP15, “Operator Challenges and Discrepancy Tracking,” Revision 15

CRDR
2807501

Section 1R17: Permanent Plant Modifications

CRDR
2790598

Section 1R19: Postmaintenance testing

Procedure
36MT-9SA02, “BOP ESFAS Load Sequencer Module Functional Test,” Revision 0

Work Order
2796881

CRDRs
2796883, 2807502

Section 1R20: Refueling and Outage Activities

Procedures
31FT-9RC01, “RCP Lube Oil Collection System Inspection,” Revision 5
40OP-9ZZ03, “Reactor Startup,” Revision 42
40OP-9ZZ07, “Plant Shutdown Mode 1 to Mode 3,” Revision 26
72ST-9RX14, “Shutdown Margin - Modes 3, 4, and 5,” Revision 9
40OP-9ZZ02, “Initial Reactor Startup following Refueling,” Revision 37

CRDRs
2790555, 2792417, 2797018, 2797373, 2793806, 2788450

Miscellaneous
Memo 294-1908-DWV, “Review of Operability Determinations Carried Over Into U2C13"

File 445-00352-WJH, “Unit 2R12 Shutdown Risk Assessment,” Revisions 1 and 2

File 445-00353-MAH, “Unit 3M12 Shutdown Risk Assessment,” Revision 0

Material Engineering Evaluation 03825, “Equivalency Evaluation to allow use of Doosan Manufacture Pressurizer Heaters at PVNGS,” Revision 0

Permits

| | | | |
|---|---|---|---|
| 112516 | 112688 | 113174 | 115532 |
| 112482 | 112887 | 114836 | 115385 |

Section 1R22: Surveillance Testing

Work Order
2703422

Procedure
73ST-9CL01, “Containment Leakage Type A and C Testing,” Revision 26

Section 4OA2: Identification and Resolution of Problems

Procedures
60DP-0QQ02, “Trend Analysis and Coding,” Revision 12
40AO-9ZZ21, “Acts of Nature,” Revision 21
73ST-9ZZ18, “Main Steam and Pressurizer Safety Valve Set Pressure Verification,” Revision 19
90DP-0IP10, “Condition Reporting,” Revision 19

CRDRs

| | | | |
|---|---|---|---|
| 2622935 | 2784544 | 2791606 | 2797246 |
| 2651750 | 2784927 | 2791966 | 2797406 |
| 2667754 | 2786245 | 2792149 | 2797530 |
| 2713099 | 2786311 | 2792326 | 2806832 |
| 2722696 | 2786378 | 2792402 | 2808389 |
| 2726522 | 2786550 | 2793816 | 2481479 |
| 2729600 | 2786643 | 2796351 | 2592898 |
| 2742173 | 2788587 | 2796508 | 2606190 |
| 2749131 | 2790091 | 2796680 | 2715709 |
| 2767879 | 2790460 | 2796891 | 2717159 |
| 2771440 | 2790736 | 2797037 | 2744078 |
| 2773643 | 2791361 | 2797158 | |
| 2784263 | | | |

Condition Report Action Items

| | | | |
|---|---|---|---|
| 2597959 | 2709394 | 2709396 | 2709399 |
| 2654847 | 2709395 | 2709397 | 34979 |
| 2696734 | | | |

LERs

| | | | |
|---|---|---|---|
| 529-2000-002-00 | 529-2001-001-00 | 530-2003-001-00 | 530-2004-003-00 |
| 529-2000-009-00 | 529-2002-001-00 | 530-2003-001-01 | |

Miscellaneous
Technical Specification 3.7.1, “Main Steam Safety Valves (MSSVs)”

PVNGS Updated FSAR (various sections), through Revision 12

VTD-D243-0007, “Instructions for Installation and Maintenance of Dresser Consolidated Safety Valves (Pub. #414),” Revision 1

Performance Improvement Department Desk Instruction, “PID Coding and Trending”

Performance Improvement Department Fourth Quarter 2004 Trend Report

Performance Improvement Department Main Category Upper Control Limit Report, October 2004 through March 2005

PVNGS Monthly Trend Reports, January 2005 through April 2005

Section 4OA5: Other Activities

Procedures
41AL-1RK1B, “Panel B01B Alarm Responses”

42AL-2RK1B, “Panel B01B Alarm Responses”

43AL-3RK1B, “Panel B01B Alarm Responses”

41ST-1ZZ02, “Inoperable Power Sources Action Statement”

42ST-2ZZ02, “Inoperable Power Sources Action Statement”

43ST-3ZZ02, “Inoperable Power Sources Action Statement”

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

70DP-0RA01, “Shutdown Risk Assessments”

40DP-9OP34, “Switchyard Administrative Control”

40DP-9AP13, “Blackout Technical Guideline”

40EP-9EO08, “Blackout”

Miscellaneous

13-ES-A15, “Station Blackout Coping Study”
Event Reporting Manual
PVNGS Draft Response to NRC Temporary Instruction 2515/156 Survey

## LIST OF ACRONYMS

| | |
|---|---|
| ASME | American Society of Mechanical Engineers |
| AFW | auxiliary feedwater |
| CAP | corrective action program |
| CEA | control element assembly |
| CFR | *Code of Federal Regulations* |
| CRDR | condition report/disposition request |
| EPRI | Electric Power Research Institute |
| ESFAS | engineered safety features actuation system |
| ET | eddy-current test |
| HPE | human performance event |
| LER | licensee event report |
| LPSI | low pressure safety injection |
| LSRO | limited senior reactor operator |
| MBA | material balance area |
| MSSVs | main steam safety valves |
| NCV | noncited violation |
| NDE | nondestructive examination |
| NRC | Nuclear Regulatory Commission |
| PID | Performance Improvement Department |
| RCS | reactor coolant system |
| SFP | spent fuel pool |
| SFHM | spent fuel handling machine |
| SIT | safety injection tank |
| SSC | structure, system, and component |
| TI | temporary instruction |
| UCL | upper control limit |
| WECC | Western Electric Coordinating Council |
| WM | work mechanism |
| WO | work order |