UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

August 9, 2004

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

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000528/2004003, 05000529/2004003, 05000530/2004003

Dear Mr. Overbeck:

On June 30, 2004, 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 July 8, 2004, with you and other members of your staff.

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

This report documents eleven NRC identified and self-revealing findings of very low safety significance (Green). Ten of these findings were determined to involve violations of NRC requirements; however, because of the very low safety significance 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. Additionally, three licensee-identified violations, which were determined to be of very low safety significance, are listed in Section 4OA7 of this report. If you contest the 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, it's 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 05000528/2004003,
05000529/2004003, and 05000530/2004003
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RIV:RI:DRP/D	RI:DRP/D	SRI:DRP/D	SRI:DRP/D	PE:DRP/D	SPE:DRP/D
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E -	**E -**	**E -**	**E -**	**/RA/**	**/RA/**
7/29/04	7/29/04	7/29/04	7/29/04	8/6/04	8/9/04

C:DRS/PSB	C:DRS/EB	C:DRS/OB	C:DRS/PEB	C:DRP/D	
MPShannon	JAClark	ATGody	LJSmith	TWPruett	
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8/6/04	8/6/04	8/9/04	8/6/04	8/9/04	

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-528, 50-529, 50-530
Licenses:	NPF-41, NPF-51, NPF-74
Report No:	05000528/2004003, 05000529/2004003, and 05000530368/2004003
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Tonopah, Arizona
Dates:	April 1 through June 30, 2004
Inspectors:	D. Carter, Health Physicist E. Crowe, Resident Inspector, Arkansas Nuclear One C. Johnson, Senior Reactor Inspector B. Henderson, Reactor Inspector J. Mateychick, Reactor Inspector T. McConnell, Reactor Inspector J. Melfi, Resident Inspector N. Salgado, Senior Resident Inspector G. Replogle, Senior Resident Inspector, Columbia Generating Station J. Tapia, Senior Reactor Inspector G. Warnick, Resident Inspector R. Azua, Project Engineer
Approved By:	Troy W. Pruett, Chief, Project Branch D Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000528/2004003, 05000529/2004003; 05000530/2004003; 04/01/04 - 06/30/04; Palo Verde Units 1, 2, and 3; Integrated Resident and Regional Report; Nonrout. Evol., Op. Evals., Post-Maint. Test., Refuel. Out., Surv. Test., Prob. Ident., Event Followup, and Other Activities.

This report covered a 3-month period of inspection by resident inspectors, five reactor inspectors, a project engineer, and a health physicist. The inspection identified 11 Green 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 4.3.2 was identified for the failure to properly maintain a check valve (siphon breaker) between the vacuum drying skid and the spent fuel pool, such that the spent fuel pool could not be inadvertently drained below 137 feet 6 inches. On May 14, 2004, the check valve failed to open and caused an inadvertent siphoning of approximately 20 gallons from the Unit 3 spent fuel pool to the cask washdown pit. Had the draindown continued, the spent fuel pool level could have decreased below 137 feet 6 inches. This issue was entered into the corrective action program as Condition Report/Disposition Request 2709518.

  The finding is greater than minor because it affected the configuration control attribute of the initiating events cornerstone. 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 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 Technical Specification 5.4.1.d was identified for the failure to ensure that hot work activities were not performed in the presence of flammable compounds. Specifically, work instructions did not require that maintenance personnel remove residual isopropyl alcohol from the main feedwater pump Train A turbine casing prior to commencing hot work activities. Consequently, a flash fire occurred when an oxygen-acetylene torch, used to preheat the metal for welding, ignited the flammable material. The issue involved human performance crosscutting aspects associated with inattention to detail by maintenance personnel. This issue was entered into the corrective action program as Condition Report/Disposition Request 2699943.

The finding is greater than minor because it could become a more significant safety concern if left uncorrected, in that a fire could ignite in an area with risk important equipment. This finding cannot be evaluated by the significance determination process because Manual Chapter 0609, "Significance Determination Process," Appendix F, "Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Findings," does not address the potential risk significance of shutdown fire protection findings. Additionally, Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," does not address fire protection findings. However, the finding is determined to be of very low safety significance by management review because the finding occurred while the unit was already in a cold shutdown condition and the finding involved equipment not necessary to maintain safe shutdown (Section 1R19).

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for the failure of the licensee to have written instructions for testing a remotely controlled submersible vehicle in the Unit 1 spent fuel pool. The vehicle became entrained in the common suction line for the spent fuel pool cooling system. At the time of the event, the unit was in refueling operations with 164 of the 241 spent fuel assemblies unloaded into the spent fuel pool. The issue involved human performance crosscutting aspects associated with poor decision making and a lack of questioning attitude by radiation protection personnel. This issue was entered into the corrective action program as Condition Report/Disposition Request 2697384.

  The finding is greater than minor because it affected the configuration control and human performance attributes of the initiating events cornerstone objective. 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 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 4OA3).

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a was identified when personnel failed to follow a maintenance procedure preceding a 12- to 24-inch heavy load drop of a 7000 pound steam generator snubber level plate inside the Unit 2 containment. The drop was due to a series of errors between the engineering contractor and rigging crews. The snubber plate was dropped in the vicinity of reactor coolant and shutdown cooling piping. This issue was entered into the corrective action program as Condition Report/Disposition Request 2639721.

The finding was greater than minor because it affects the equipment performance and human performance attributes of the initiating events cornerstone objective to limit the likelihood of events that challenge safety functions during shutdown conditions. Using Manual Chapter 0609, "Significance Determination Process," Appendix G, "Shutdown Operations Significance Determination Process," the senior reactor analyst concluded that this finding did not significantly increase the likelihood of losing the residual heat removal function and did not significantly increase the likelihood that systems that could mitigate a loss of residual heat removal function would be degraded. Therefore, this finding is of very low safety significance (Section 4OA5).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to identify the root cause of spent fuel pool inventory loss events and implement corrective actions to preclude recurrence. Specifically, the improper positioning of a fuel pool cleanup suction valve and inadequate level monitoring resulted in three losses of spent fuel pool inventory events. This finding involves problem identification and resolution crosscutting aspects associated with the failure to identify root causes and implement corrective actions. The issue also involved human performance crosscutting aspects associated with mispositioned valves and awareness of plant conditions by operations personnel. This issue was entered into the corrective action program as Condition Report/Disposition Request 2599869.

  The finding is greater than minor because it affected the configuration control and human performance attributes of the initiating events cornerstone objective. 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 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 4OA5).

Cornerstone: Mitigating Systems

- SLIV. A Severity Level IV noncited violation of Technical Specification 3.3.11 was identified for the failure to include the resistance temperature detectors in the channel calibration for the shutdown cooling heat exchanger temperature instruments. Specifically, prior to the implementation of Improved Technical Specifications, the licensee did not perform testing of the resistance temperature detectors. Following the implementation of Improved Technical Specifications, the licensee did not perform an in-place qualitative assessment of the resistance temperature detectors' behavior. This issue was entered into the corrective action program as Condition Report/Disposition Request 280178.

  The failure to perform a complete shutdown cooling heat exchanger temperature loop channel calibration is determined to have greater than minor significance because the licensee's failure to report the condition impacted the NRC's ability to perform it's regulatory function. Therefore, this finding was considered applicable to traditional enforcement. Although the significance determination process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, the finding can be assessed using the significance determination process. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," this finding is determined to be of very low safety significance because it only affected the mitigating system cornerstone and the resistance temperature detectors were found to be within calibration (Section 4OA2).

- Green. The inspectors identified a noncited violation of Technical Specification 5.2.2.d for the failure of authorized individuals to review monthly overtime reports to ensure that excessive hours have not been assigned. Specifically, following the implementation of an electronic reporting system in 2001, the licensee did not ensure that all managers continued to receive and approve the Excess Hours Report.

  The finding is greater than minor because if left uncorrected it could become a more significant safety concern, in that exceeding the NRC Generic Letter 82-02, "Nuclear Power Plant Staff Working Hours," guidelines for overtime limits is a contributor to worker fatigue. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," this finding is determined to be of very low safety significance because there were no known actual adverse plant or equipment conditions that could be attributed to worker fatigue (Section 4OA2).

Cornerstone: Barrier Integrity

- SLIV. A Severity Level IV noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to correct a nonconforming condition in a timely manner. Specifically, since June 2001, the licensee discontinued implementation of required Technical Specification surveillance testing for the containment purge valves by declaring the valves inoperable and installing blind flanges. This issue was entered into the corrective action program as Condition Report/Disposition Request 2711167.

The finding is greater than minor because the licensee's failure to submit a license amendment to correct the nonconforming condition impacted the NRC's ability to perform its regulatory function. Therefore, this finding was considered applicable to traditional enforcement. Although the significance determination process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, the finding can be assessed using the significance determination process. 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 affected the barrier integrity cornerstone and the installation of blind flanges adequately maintained containment integrity (Section 1R15).

- <u>Green</u>. A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to correct a degraded refueling machine equipment condition that could have impacted the ability to safely handle fuel. Specifically, refueling personnel continued to move spent fuel even though they had determined that the refueling machine sprag brake had failed. The issue involved human performance crosscutting aspects associated with poor decision making and a lack of questioning attitude by refueling personnel. This issue was entered into the corrective action program as Condition Report/Disposition Request 2704331.

  The finding is greater than minor since it could become a more significant safety concern if left uncorrected in that continuing core alterations using degraded equipment impacts the ability to safely handle spent fuel and increases the likelihood of a fuel handling accident. Using the Phase 1 worksheets in Manual Chapter 0609, "Significance Determination Process," this finding is determined to have very low safety significance because it only affects the barrier integrity cornerstone and was a deficiency that did not result in the actual degradation of spent fuel (Section 1R20).

- <u>Green</u>. A self-revealing finding was identified when a pressurizer level transient above Technical Specification limits occurred. Specifically, simultaneous testing of the atmospheric dump valve and boron injection systems resulted in a loss of letdown event on high regenerative heat exchanger temperature. The letdown event occurred because operations personnel were using a single charging pump for the boron injection test and using excess letdown to accommodate a plant heatup following atmospheric dump valve testing. The combination of activities resulted in pressurizer level exceeding the Technical Specification limit of 56 percent. The issue involved human performance crosscutting aspects associated with operator decision making, questioning attitude, awareness of plant conditions, and communications between personnel performing concurrent evolutions. This issue was entered into the corrective action program as Condition Report/Disposition Request 2707290.

The finding is greater than minor because it is associated with the equipment performance attribute of the barrier integrity cornerstone and affects the cornerstone objective of protecting the reactor coolant system barrier from radionuclide releases caused by accidents or 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 barrier integrity cornerstone and was a deficiency that did not result in the actual degradation of the reactor coolant system barrier (Section 1R22).

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified for the failure to secure a main steam line pipe whip restraint inside the Unit 2 containment in accordance with design drawings. Specifically, the pipe whip restraint was missing four ½-inch diameter nuts from the embedded anchor bolts. This issue was entered into the corrective action program as Condition Report/Disposition Request 2643347.

  The finding is greater than minor since it is associated with the equipment performance attribute of the barrier integrity cornerstone and affects the cornerstone objective of protecting the containment barrier from radionuclide releases caused by accidents or events. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," this finding is determined to have very low safety significance because it did not represent an actual open pathway in the physical integrity of the reactor containment and did not represent an actual reduction of the atmospheric pressure control function of the reactor containment (Section 4OA5).

B. Licensee-Identified Violations

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program (Section 4OA7).

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full power until April 3, 2004, when the unit was shut down for Refueling Outage 1R11. Following reactor startup on May 8, 2004, the reactor was manually tripped from Mode 2 following slipping of a control element assembly (CEA) approximately 6 inches into the core. Repairs were completed and the unit returned to essentially full power on May 14, 2004. On June 14, 2004, Unit 1 experienced a reactor trip after a loss of offsite power (LOOP). Following the evaluation of the event and impacted systems, a reactor startup was completed and the unit was returned to essentially full power on June 20, 2004. On June 21, 2004, Unit 1 experienced a reactor power cutback to 55 percent power due to the loss of main feedwater (MFW) pump Train B on low suction pressure. Repairs were completed and the unit was returned to essentially full power on June 23, 2004, and remained there for the duration of the inspection period.

Unit 2 operated at full power until June 14, 2004, when the unit experienced a reactor trip after a LOOP. Following the evaluation of the event and impacted systems, a reactor startup was completed, and the unit was returned to essentially full power on June 20, 2004, where it remained for the duration of the inspection period.

Unit 3 operated at full power operation until May 11, 2004, when a leak developed on the Condenser C vent line, requiring a reduction in power to approximately 40 percent. Repairs were completed and the unit was returned to essentially full power on May 15, 2004. On June 7, 2004, a reactor power cutback followed by an automatic reactor scram occurred when an electrohydraulic control cabinet electrical malfunction resulted in the closure and subsequent reopening of three combined intercept valves. Reactor startup was completed on June 9, 2004, and Mode 1 entered on June 10. At approximately 12 percent reactor thermal power, the main turbine was manually tripped following unexpected combined intercept valve movement. On June 11, 2004, Mode 2 was re-entered and the reactor was downpowered to approximately 3 percent reactor thermal power while additional troubleshooting was performed. The licensee determined that a backup speed probe had an intermittent failure. Once repairs were completed, a reactor startup was initiated and the unit was returned to essentially full power on June 13, 2004. On June 14, 2004, Unit 3 experienced a reactor trip after a LOOP. Following the evaluation of the event and impacted systems, a reactor startup was completed and the unit was returned to essentially full power on June 21, 2004, where it remained for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors completed a partial walkdown of the three systems listed below to verify proper equipment alignment. This inspection included a review of the applicable plant procedures, plant drawings, outstanding modifications, work orders (WOs), and condition report/disposition requests (CRDRs). The inspectors verified the following: all valves were properly aligned; there was no leakage that could affect operability; electrical power was available as required; major system components were properly labeled, lubricated, and cooled; and hangers and supports were correctly installed and functional.

- April 6, 2004, shutdown cooling (SDC) Train B during midloop operations (Unit 1)
- April 9, 2004, emergency diesel generator Train B (Unit 1)
- April 28, 2004, emergency diesel generator Train A (Unit 1)

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors conducted tours of the five areas listed below that are important to reactor safety and referenced in the Pre-fire Strategies Manual to evaluate conditions related to licensee control of transient combustibles and ignition sources; the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and the fire barriers used to prevent damage from propagation of potential fires.

- April 8, 2004, containment building all accessible areas (Unit 1)
- April 20, 2004, fire pump house (Units 1, 2, and 3)
- June 2, 2004, fuel building all elevations (Unit 1)
- June 3, 2004, fuel building all elevations (Unit 2)
- June 4, 2004, fuel building all elevations (Unit 3)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

During the Unit 1 outage, licensee personnel conducted an inspection of the Train A essential cooling water heat exchanger. The inspectors reviewed test and analysis results for the Train A essential cooling water heat exchanger. Using Procedure 70TI-9EW01, "Thermal Performance Testing of Essential Cooling Water Heat Exchangers," Revision 4, heat exchanger data was collected on April 3, 2004. Additional data was collected on May 5, 2004, after the licensee cleaned the heat exchanger. The data was analyzed using Procedure 73DP-9ZZ10, "Guidelines for Heat Exchanger Thermal Performance Analysis," Revision 4. The inspectors' review was conducted to determine if the test acceptance criteria and results appropriately considered the differences between testing and design conditions and if the results were appropriately measured against pre-established acceptance criteria.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

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

Procedure 71111.08 requires the review of a minimum sample of five nondestructive examination activities of at least three different types. The inspector witnessed the performance of five volumetric examinations, nine surface examinations, and one visual examination. This sample of 15 nondestructive examination activities of three types is listed in the enclosed attachment.

For each of the nondestructive examination activities reviewed, the inspector verified that the examinations were performed in accordance with American Society of Mechanical Engineers (ASME) Code requirements.

During the review of each examination, the inspector verified that appropriate nondestructive examination procedures were used, that examinations and conditions were as specified in the procedure, and that test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspector also reviewed documentation to verify that indications revealed by the examinations were dispositioned in accordance with the ASME Code specified acceptance standards.

The inspector verified the certifications of approximately five nondestructive examination personnel (Lambert-MacGill-Thomas, Inc.) observed performing examinations or identified during review of completed examination packages.

The inspection procedure requires review of one or two examinations from the previous outage with recordable indications that were accepted for continued service to ensure that the disposition was done in accordance with the ASME Code. The inspector selected several recordable indications that required evaluation during the last outage. The licensee evaluated the indications in accordance with the ASME Code requirements. These indications did not exceed any code requirements. Corrective actions reviewed by the inspector were appropriate.

If the licensee completed welding on the pressure boundary for Class 1 or 2 systems since the beginning of the previous outage, the procedure requires verification that acceptance and preservice examinations were done in accordance with the ASME Code for one to three welds. The inspector reviewed the welding activity associated with the replacement of Class 2 piping on the containment spray system discharge during the current outage (WO 2590635).

The procedure also requires verification that one or two ASME Code Section XI repairs or replacements meet code requirements. The inspector reviewed and observed one ASME Code, Section XI, replacement activity associated with the replacement of Class 2 piping on the containment spray system discharge line. The inspector verified that the repair or replacement activities were in accordance with Section XI requirements.

b. Findings

No findings of significance were identified.

2. Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified 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 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 inspector observed the performance of a pressure test performed on Steam Generator Tube 148-87. The test was performed in accordance with the Westinghouse in-situ test procedure. Procedures reviewed were appropriate for the test required.

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 inspector verified that the licensee's prediction of flaws detected this outage were consistent with the actual flaws identified through their flaw predication program.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet Technical Specification requirements, Electric Power Research Institute guidelines, and commitments made to the NRC. The inspector reviewed the steam generator tube eddy current test scope and expansion criteria and found no deficiencies.

The inspection procedure required confirmation that the licensee inspected all areas of potential degradation, especially areas which were known to represent potential eddy current test challenges (e.g., top-of-tube sheet, tube support plates, batwing area, and U-bends). The inspector confirmed whether all known areas of potential degradation, including eddy current test challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure also required confirmation of adherence to the Technical Specification plugging limit. The inspection procedure required determination of 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 inspector confirmed that the licensee was adhering to these specifications.

If steam generator leakage greater that 3 gallons per day was identified during operations or during postshutdown visual inspections of the tube sheet face, the inspection procedure required verification that the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspector determined that leakage greater than 3 gallons per day did not exist.

The inspection procedure required confirmation that the eddy current test 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 inspector observed portions of all eddy current tests performed. During these examinations, the inspector 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 inspection procedure requires confirmation of the licensee's corrective action if loose parts or foreign material on the secondary side is identified. The inspector reviewed the licensee's foreign object search and retrieval summary report for the Refueling Outage 1R11 steam generator secondary side. This report indicates that the licensee is aggressively tracking and identifying any loose parts within the steam generators' secondary side and implementing appropriate corrective actions when needed.

Finally, the inspection procedure specified review of one to five samples of eddy current test data if questions arose regarding the adequacy of eddy current test data analyses. The inspector did not identify any results where eddy current test data analyses adequacy was questionable.

b. Findings

No findings of significance were identified.

3. Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed approximately seven inservice inspection related CRDRs issued during the current and past refueling outages. The review served to verify that the licensee’s corrective action process was being correctly utilized to identify conditions adverse to quality and that those conditions were being adequately evaluated, corrected, and trended. The inspector also concluded that corrective actions were being appropriately addressed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On May 20, 2003, the inspectors observed operations crew performance during evaluated simulator Scenario SES-10-A-00, "Steam Generator Tube Leak." The inspectors evaluated the simulator scenario, the crew performance, and the evaluator critique sessions conducted following the completion of the simulator scenario. Additionally, the inspectors compared simulator board configurations with actual control room board configuration for consistency.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

For the two failures listed below, the inspectors verified the licensee's appropriate handling of structure, system, and component performance or condition problems; reviewed the use of industry operating experience for establishing preventive maintenance programs; and verified that licensee personnel properly implemented the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants":

- Failure of high pressure safety injection Valve 3JSIAUV0627 to stroke closed on demand, documented in CRDR 2682409, dated February 10, 2004, (Unit 3)

- Low pressure indication for main steam isolation Valve MSIV-180 Train B accumulator, documented in CRDR 2693900 (Unit 2)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Throughout this inspection period, the inspectors reviewed daily and weekly work schedules to determine when risk significant activities were scheduled. The inspectors reviewed risk evaluations and overall plant configuration control for four selected activities to verify compliance with Procedure 30DP-9MT03, "Assessment and Management of Risk When Performing Maintenance in Modes 1 - 4," Revision 8. The inspectors discussed emergent work issues with work control personnel and reviewed the potential risk impact of these activities to verify that the work was adequately planned, controlled, and executed.

- April 24, 2004, evaluated the licensee's assessment associated with moving the upper guide structure lift rig to the core support stand for CEA 16 changeout with fuel in the vessel. CEA Extension Shaft 16 was stuck into its respective self- latching mechanism (Unit 1)

- June 2, 2004; evaluated the licensee's assessment associated with Startup Transformer AENANX03 being unable to be electrically isolated due to the failure of the 525 kV crossover breaker between Startup Transformer X03 and Devers MAN-PL-995 (Unit 3)

- June 3, 2004; troubleshooting position indication loss for high pressure safety injection discharge Valve 2JSIAHV0698 per WO 2713742 (Unit 2)

- June 18, 2004; evaluated the licensee's controls for maintaining voltage limits for offsite power when Unit 1 paralleled the main generator to the grid (Unit 1)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors observed the following seven nonroutine evolutions to verify that they were conducted in accordance with licensee procedures and Technical Specifications:

- On April 19, 2004, an underground fire main ruptured, spilling approximately 278,000 gallons of water. The rupture occurred at the plant northeast corner of the Unit 3 essential spray pond. The rupture was isolated in approximately 45 minutes by the fire department.

  The inspectors evaluated that the actions taken by operations and the fire department personnel were in accordance with licensee procedures and the Technical Requirements Manual. The inspectors discussed the plant response to the event with the control room operators, fire department personnel, and plant management. This event was documented in CRDR 2700170.

- On May 8, 2004, Unit 1 reactor operators manually tripped the reactor when CEA 89 slipped approximately 6 inches while conducting low power physics testing following Refueling Outage 11.

  The inspectors responded to the control room to evaluate the plant conditions and operator performance. The inspectors performed a control board walkdown to verify all safety equipment responded as required. The inspectors discussed the plant response to the event with the control room operators and plant management. This Unit 1 event was documented in CRDR 2707423.

- On May 11, 2004, Unit 3 developed a leak on the Condenser C vent line (Circulating Water Line 3PCWNL031 near Valve CWNV081), requiring a reduction in power to approximately 40 percent.

  The inspectors responded to the control room to evaluate the plant conditions and operator performance. The inspectors performed a control board walkdown to verify all safety equipment responded as required. The inspectors discussed the plant response to the event with the control room operators and plant management. This Unit 3 event was documented in CRDR 2707578.

- On May 14, 2004, while performing vacuum drying on Canister 16 in the Unit 3 cask load pit, an unintended transfer of approximately 20 gallons of water occurred from the spent fuel pool (SFP) to the cask load pit via the cask drainage piping. The licensee entered abnormal operating Procedure 40AO-9ZZ23, "Loss of SFP Level or Cooling," Revision 7, for this event.

  The inspectors responded to this event, walked down the area, and reviewed drawings of the installation cask drainage piping. The inspectors discussed the event with the control room operators and plant management. This event is documented in CRDR 2709518.

- On June 7, 2004, Unit 3 had a reactor power cutback and reactor trip due to a problem with the electrohydraulic control system.

  The inspectors responded to the control room to evaluate the plant conditions and operator performance. The inspectors performed a control board walkdown to verify all safety equipment responded as required. The inspectors discussed the plant response to the event with the control room operators and plant management. This Unit 3 event was documented in CRDR 2714544.

- On June 14, 2004, the inspectors responded to the control rooms to evaluate the plant conditions and operator performance following the LOOP and reactor trips on Units 1, 2, and 3. Units 1 and 3 declared a notice of unusual event for the LOOP, and Unit 2 declared an alert, due to emergency diesel generator Train A not maintaining voltage. The inspectors responded to Unit 2 and the technical support center. Visiting reactor inspectors responded to Units 1 and 3. The inspectors discussed the plant response to the event with the control room operators and plant management.

  On June 14, 2004, an Augmented Inspection Team was chartered to evaluate this event. The results will be documented in NRC Augmented Inspection Report 05000528; 05000529; 05000530/2004012.

- On June 21, 2004, Unit 1 had a reactor power cutback to approximately 55 percent power, due to a trip of Condensate Pump B.

  The inspectors responded to the control room to evaluate the plant conditions and operator performance. The inspectors performed a control board walkdown to verify all equipment responded as required. The inspectors discussed the plant response to the event with the control room operators and plant management. This Unit 1 event was documented in CRDR 2717298.

b. Findings

Introduction. A Green self-revealing noncited violation of Technical Specification 4.3.2 was identified for the failure to maintain a check valve (siphon breaker) between the vacuum drying skid and the SFP. On May 14, 2004, the check valve failed to open and caused an inadvertent siphoning of approximately 20 gallons from the Unit 3 SFP to the cask washdown pit.

Description. On May 14, 2004, the licensee performed vacuum drying on Canister 16 in the Unit 3 cask load pit. The vacuum drying process had been completed, and the licensee was preparing to perform a rate-of-rise test on the cask to verify all water had been removed. After manipulating valves and stopping the vacuum pump, licensee personnel identified water leaking from a crack on the vacuum pump discharge muffler. The licensee attempted to stop the leak by opening a drain valve on the discharge of the pump. After approximately 30 seconds of draining this discharge line, the licensee recognized that the leakage was water that was being siphoned from the SFP due to a check valve that failed to open. The licensee took immediate actions to remove the check valve in order to establish a vent path to terminate the water transfer from the SFP.

Through interviews with the licensee the inspectors were informed that the check valve had been installed in January 2004. The inspectors noted that the check valve was not tested or in a preventive maintenance program.

Analysis. The inspectors identified a performance deficiency for the failure to perform postinstallation testing that would demonstrate the operability of the check valve, thus preventing the possibility of an inadvertent draining of the SFP. The finding is greater than minor because it affected the configuration control attribute of the initiating events cornerstone. 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 management review because radiation shielding was provided by the SFP water level, the SFP cooling system and fuel building ventilation systems were available, and there were multiple sources of makeup water.

Enforcement. Technical Specification 4.3.2 required that the spent fuel storage pool be designed and maintained to prevent inadvertent draining of the pool below 137 feet 6 inches. Contrary to the above, a check valve on a vacuum drying skid connected to the SFP was not maintained to assure that it would prevent inadvertent draining of the SFP. On May 14, 2004, this check valve failed to open and caused an inadvertent draining of approximately 20 gallons of water from the Unit 3 SFP to the cask washdown pit. Had operators not intervened, the failure of this check valve could have potentially caused an inadvertent draining of the SFP to elevation 127 feet 3 inches (below elevation 137 feet 6 inches required by Technical Specifications). Because the finding was of very low significance and has been entered into the licensee's corrective action program as CRDR 2709518, this finding is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000530/2004003-01, "Spent Fuel Pool Water Siphon due to Check Valve Failure."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors evaluated the six operability determinations listed below for technical adequacy and assessed the impact of the condition on continued plant operation. Additionally, the inspectors reviewed Technical Specification entries, CRDRs, and equipment issues to verify that operability of plant structures, systems, and components was maintained or that Technical Specification actions were properly entered.

- May 5, 2004; resolution of equipment failures following emergency diesel generator Train A outage maintenance and associated operability justifications documented in CRDRs 2703945 and 2705929 (Unit 1)

- May 11, 2004; assessed the resolution of azimuthal tilt and increase of core operating limits report limit documented in CRDR 2707812 (Unit 1)

- April 15, 2004; operability evaluation for Snubber 2SI123H001 installed on the containment spray/low pressure safety injection system discharge piping that was identified as potentially locked-up on March 30, 2004, as documented in CRDRs 2693663 and 2704218 (Unit 2)

- Reviewed Technical Specification Component Condition Record 2375153, "LCO 3.6.3 Containment Isolation Valves SR 3.6.3.6 Not Performed Adequately for CP 2B and 3A," and the licensee's overall response to the degraded and nonconforming condition (Units 1, 2, and 3)

- April 27, 2004; assessed the impact of potential unqualified coatings being identified in at least five areas at the 80-foot elevation of containment (Unit 1)

- June 25, 2004; assessed available thrust for motor-operated Valve 2JSGNHV1143 was less than minimum administrative limit as documented in CRDR 2639681 (Unit 2)

b. Findings

Introduction. A Severity Level IV noncited violation was identified for failure to correct a nonconforming condition in a timely manner. The nonconformance involved long-term actions taken to compensate for containment purge isolation valve design deficiencies.

Description. In March 2001, the licensee determined that the 42-inch containment purge isolation Valve CP-UV-2A/3B, had unreliable seals against containment pressure and declared the valves inoperable. On June 15, 2001, the licensee developed an interim strategy for containment purge Penetrations 56 and 57 due to the inability to satisfy Technical Specification Surveillance Requirement 3.6.3.6. The interim strategy involved declaring the inboard and outboard valves inoperable and installing blind flanges to comply with the required actions of Technical Specification 3.6.3, Condition D, in Modes 1-4. This strategy discontinued the performance of leak rate testing of the valves and enable continued operations with the installation of blind flanges on Units 1, 2, and 3. On June 18, 2002, the licensee approved a long-term strategy to make the 42-inch containment purge penetration blind flanges part of the permanent plant configuration.

Technical Specification Bases 3.0.2 states, in part, that intentional entry into ACTIONS should not be made for operational convenience. The inspectors determined that the interim strategy adopted by the licensee inappropriately used Technical Specification actions. Further, the inspectors observed that the licensee planned to use the actions required by Technical Specification 3.6.3, Condition D, to continue plant operations until implementation of a permanent modification in 2005 and 2006. The inspectors concluded that the licensee's schedule to correct the nonconforming condition through permanent plant modification did not meet NRC guidelines. Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," states, in part, that the NRC expects time frames longer that the next refueling outage to be explicitly justified by the licensee as part of the deficiency tracking documentation. The inspectors concluded that a permanent plant modification should have been implemented at the first available opportunity following identification of the degraded and nonconforming condition. This conclusion is based, in part, on the lack of justification for intentional entry into the actions of Technical Specification 3.6.3, Condition D, during Modes 1-4. Timely correction of the nonconforming condition would have identified the need for NRC review of a license amendment through 10 CFR 50.59(c)(1).

Analysis. The failure to correct the nonconforming condition in a timely manner through permanent plant modification is determined to have more than minor significance because the licensee's failure to submit a license amendment impacted the NRC's ability to perform its regulatory function. This finding is associated with the barrier integrity cornerstone. This finding was considered applicable to traditional enforcement. Although the significance determination process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, the finding can be assessed using the significance determination process. 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 affected the barrier integrity cornerstone and the installation of blind flanges adequately maintained containment integrity.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that conditions adverse to quality be promptly identified and corrected. Contrary to the above, the licensee did not correct a condition adverse to quality in a timely manner. Specifically, the licensee failed to correct the 42-inch containment purge penetration nonconforming condition at the first available opportunity. In place of promptly correcting the condition, the licensee elected to implement the actions of Technical Specification 3.6.3, Condition D, in Modes 1-4 instead of restoring the purge valves to an operable condition. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CRDR 2711167.
NCV 05000528; 05000529; 05000530/2004003-02, "Containment Purge Penetration NonConformance."

1R17 Permanent Plant Modifications (71111.17A)

Fillet Weld Buildup on Small Drain Lines on the Safety Injection Suction Line Train B

a. Inspection Scope

The inspectors reviewed the modifications to the welds on Unit 1 safety injection suction line Train B via Design Master WO 2692447 and Design Implementation WO 2695338 to verify that it was being performed in accordance with regulatory requirements and plant procedures. The inspectors interviewed the licensee personnel installing the modification as to their understanding of the modification package and observed work in progress. The inspectors also observed portions of modification work to verify that: (1) the work package was at the work site, (2) transient combustible material was appropriately controlled, (3) construction material was appropriately staged, and (4) construction debris was kept to a minimum.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed or evaluated the results from the following eight postmaintenance tests to determine whether the test adequately confirmed equipment operability. The inspectors also verified that postmaintenance tests satisfied the requirements of Procedure 30DP-9WP04, "Postmaintenance Testing Development," Revision 13.

- April 7, 2004; cleaning of the emergency diesel generator Train B fuel oil storage tank via WO 2439603 (Unit 1)

- April 14, 2004; removal of an electrical boot installed on Valve 1-SIB-UV-0626 identified during the performance of Preventive Maintenance Task 2612939 (Unit 1)

- April 17, 2004; welding repairs on the MFW pump Train A turbine casing drains per WOs 2609675 and 2699452 (Unit 1)

- April 19, 2004; troubleshooting and resolution of low pressure safety injection Pump B abnormal noise identified during performance of Procedure 40OP-9CH12, "Refueling Water Tank Operations," Section 8, Revision 18 (Unit 2)

- May 7, 2004; troubleshooting of sticky latching mechanism for the upper gripper coil on CEA 89 per WO 2706748 (Unit 1)

- June 3, 2004; high pressure safety injection flow verification using Procedure 73ST-9SI10, "HPSI A Inservice Test," Revision 29, per WO 2592003 (Unit 2)

- June 8, 2004; troubleshooting of indication of a low lube oil pressure trip on emergency diesel generator Train A per WO 2714674 (Unit 3)

- June 15-16; troubleshooting of the failure of emergency diesel generator Train A to maintain voltage per WO 2715735 (Unit 2)

b. Findings

1. MFW Pump Fire Flash

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.4.1.d for the failure to remove residual alcohol from the MFW pump Train A turbine casing prior to commencing hot work activities. Consequently, a flash fire occurred when a oxy-acetylene torch was used to preheat the metal for welding.

Description. On April 17, 2004, while performing planned corrective maintenance (weld buildup) on the drains on the MFW pump Train A turbine casing, an unexpected flash fire occurred. Isopropyl alcohol was used during the final cleaning process for the turbine casing. However, as part of the maintenance activity, maintenance personnel did not adequately remove the residual alcohol prior to the commencement of the hot work. The flash fire occurred while preheating the turbine casing metal with an oxy-acetylene torch and resulted in two mechanics receiving burns to their exposed facial areas.

The licensee conducted a site-wide standdown on the event and performed an investigation documented in CRDR 2699943. The licensee determined that the maintenance personnel had not been adequately informed of personnel safety hazards, as required by Procedure 30DP-9WP02, "Work Document Development and Control," Revision 34. Also, the controlling WOs 2609675 and 2699452 did not adequately identify appropriate cleaning instructions to ensure that residual alcohol from the MFW pump turbine casing was removed prior to preheating the metal.

Analysis. The finding is greater than minor because it could become a more significant safety concern if left uncorrected, in that a fire could ignite in an area with risk important equipment. This finding is associated with the initiating events cornerstone. This finding cannot be evaluated by the significance determination process. Manual Chapter 0609, "Significance Determination Process," Appendix F, "Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Findings," does not address the potential risk significance of shutdown fire protection findings. Additionally, Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," does not address fire protection findings. However, the finding is determined to be of very low safety significance by management review because the finding occurred while the unit was already in a cold shutdown condition and the finding involved equipment not necessary to maintain safe shutdown.

Enforcement. Technical Specification 5.4.1.d requires, in part, that written procedures be established, implemented, and maintained for fire protection program implementation. Fire protection program Procedure 14DP-0FP36, "Hot Work Permit," Revision 9, step 3.1.1.2, required, in part, that hot work not be performed in areas where the presence of other flammable compounds creates a hazard. Contrary to this requirement, isopropyl alcohol, a flammable compound, was not adequately cleaned from the surfaces of the MFW pump Train A turbine casing. Consequently, a fire occurred when an oxy-cetylene torch was used to preheat the metal for welding. Because the finding is of very low safety significance and has been entered into the corrective action program as CRDR 2699943, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2004003-03, "Fire That Occurred During Welding Activities on MFW Pump Turbine Train A."

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the licensee's Unit 1 Refueling Outage 1R11 shutdown risk assessment to confirm that the licensee had appropriately considered risk, industry experience, and a previous site-specific problem in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the attachment.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the shutdown risk assessment for key safety functions and compliance with the applicable Technical Specifications when taking equipment out of service

- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing

- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error

- Controls over the status and configuration of electrical systems to ensure that Technical Specification and outage safety plan requirements were met, and controls over switchyard activities

- Monitoring of decay heat removal processes

- Controls to ensure that outage work was not impacting the ability of the operators to operate the SFP cooling system

- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss

- Controls over activities that could affect reactivity

- Maintenance of secondary containment as required by Technical Specification

- Refueling activities

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system (ECCS) suction strainers, and reactor physics testing

- Licensee identification and resolution of problems related to refueling outage activities

b. <u>Findings</u>

<u>Degraded Refueling Machine</u>

<u>Introduction</u>. A Green noncited violation was identified for failing to correct a degraded refueling machine equipment condition that could have impacted the ability to safely

handle fuel.

Description. On April 27, 2004, the inspectors questioned the licensee's decision to continue core alterations following the identification of a degraded equipment condition associated with the refueling machine hoist. The refueling machine driver noticed that the sprag brake was not functioning properly while lowering a fuel assembly into the reactor vessel. The sprag brake functions to slow the downward motion if the speed exceeds that of the input mode, either electrical motor or manual crank shaft, used to lower the assembly. The licensee decided to electrically raise and lower the fuel assembly to determine if the sprag brake was functioning properly. Following this troubleshooting evolution, which confirmed that the sprag brake was degraded, the licensee manually lowered the fuel assembly into the appropriate core location in accordance with contingencies provided in Procedure 78OP-9FX01, "Refueling Machine Operations," Revision 17, Appendix G, "Manual Operation of the Refueling Machine."

The licensee decided to continue core alterations with the degraded refueling machine hoist to raise a second fuel assembly already located in the vessel, rotate the assembly to a different orientation, and then manually lower it back into the reactor vessel. The inspectors identified that the licensee inappropriately implemented contingencies in Procedure 78OP-9FX01, Appendix K, "Action Plan for Movement of a Difficult Assembly," to operate the refueling machine hoist in manual mode. Step 3.1.5 of this procedure states, "The refueling machine hoist shall not be operated manually except to place a fuel assembly in a safe condition, or for testing, maintenance, or operations involving an assembly which is not seating, lowering, or raising correctly." The licensee justified manual hoist operation based on the need to use Appendix K when the assembly was initially seated in the vessel in a rotated position, even though a "difficult assembly" condition did not actually exist. Furthermore, operators believed that the degraded brake condition only affected electric motor operation and failed to recognize that the degraded condition also impacted manual hoist operation. The licensee suspended further core alterations following movement of the second fuel assembly until refueling machine sprag brake repairs were completed.

Analysis. The finding is greater than minor since it would become a more significant safety concern if left uncorrected, in that continuing core alterations using degraded equipment impacts the ability to safely handle fuel and increases the probability of a fuel handling accident. This finding is associated with the barrier integrity cornerstone. Using the Phase 1 worksheets in Manual Chapter 0609, "Significance Determination Process," the finding was determined to have very low safety significance because it only affects the barrier integrity cornerstone and was a deficiency that did not result in the actual degradation of the fuel cladding barrier. The finding involved human performance crosscutting aspects associated with poor decision making and a lack of questioning attitude by operations and refueling personnel.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that conditions adverse to quality be promptly identified and corrected. Contrary to the above, the licensee did not promptly correct a condition adverse to quality. Specifically, the licensee identified a degraded equipment condition associated with the refueling machine sprag brake and continued core alterations prior to correcting the degraded condition. Because the finding is of very low safety significance and has been entered into the corrective action program as CRDR 2704331, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2004003-04, "Core Alterations with Degraded Refueling Machine."

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

Applicable test data was reviewed to verify whether the licensee met Technical Specification, Updated Final Safety Analysis Report, and licensee procedure requirements. The inspectors verified that testing effectively demonstrated that systems were operationally ready and capable of performing their intended safety functions and that identified problems were entered into the corrective action program for resolution. The inspectors observed the performance of and reviewed documentation for the following five surveillance tests:

- On April 12, 2004, performance of Procedure 73ST-9LC01-1, "Containment Leakage Type "B" and "C" Testing," on Penetration 22 (Unit 1)
- On April 15-16, 2004, performance of Procedure 73ST-9DG02, "Class 1E Diesel Generator and Integrated Safeguards Test Train B," Revision 8, Section 8.5, "DG-B 24 Hour Continuous Load Test/100% Load Rejection/DG-B Hot Start" (Unit 1)
- On May 7, 2004, performance of Procedure 73TI-9SG03, "ADV 30% Partial Stroke Test," Revision 5 (Unit 1)
- On June 9, 2004, performance of Procedure 36ST-9SI05, "SDC Interlocks Loop & Alarm Calibration," Appendix G, Revision 4 (Unit 3)
- On June 1, 2004, Procedure 73ST-9X113, "Train A HPSI Injection and Miscellaneous SI Valves - Inservice Test," Revision 18 (Unit 2)

b. Findings

Introduction. A Green self-revealing finding was identified when a pressurizer level transient above Technical Specification limits occurred in Mode 3 while performing simultaneous evolutions that affected reactor coolant system (RCS) inventory.

Description. On May 7, 2004, the licensee was performing two surveillances, Procedure 73TI-9SG03, "ADV 30% Partial Stroke Test," Revision 5, and Procedure 40ST-9CH04, "Boron Injection Flow Test," Revision 1. Simultaneous performance of these evolutions caused a loss of letdown due to high regenerative heat exchanger outlet temperature. This condition occurred due to single charging pump operation per Procedure 40ST-9CH04, combined with increased letdown flow to accommodate the RCS heatup following atmospheric dump valve (ADV) partial stroke testing. Operators implemented Procedure 40AO-9ZZ05, "Loss of Letdown," Revision 12, and restored letdown within 2 minutes. Subsequently, a pressurizer level transient occurred to a level greater than 56 percent, requiring entry into Technical Specification 3.4.9, Condition A, for 23 minutes.

Analysis. The inspectors determined that the finding is a performance deficiency because operators elected to perform a combination of surveillance tests that caused a loss of letdown and pressurizer level transient above Technical Specification limits. This finding is associated with the barrier integrity cornerstone. The upper pressurizer level limit is to ensure that enough steam space volume is available to accommodate insurges from anticipated transients and ensures steam passage through the safety relief valves if called upon. The finding is more than minor since it is associated with the equipment performance attribute of the barrier integrity cornerstone and affects the cornerstone objective of protecting the RCS barrier from radionuclide releases caused by accidents or events. Using the Phase 1 worksheets in Manual Chapter 0609, "Significance Determination Process," the finding is determined to have very low safety significance because it only affects the barrier integrity cornerstone and was a deficiency that did not result in the actual degradation of the RCS barrier. This issue involves human performance crosscutting aspects associated with poor decision making, questioning attitude, awareness of plant conditions, and communications between personnel performing concurrent evolutions.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because surveillance procedures were followed and pressurizer level was restored within the time required by Technical Specifications. This finding has been entered into the corrective action program as CRDR 2707290. FIN 05000528/2004003-05, "Pressurizer Level Transient Above Technical Specification Limits."

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the two temporary modifications and associated 10 CFR 50.59 screening evaluations listed below. The inspectors reviewed these against the system design basis documentation and verified that the modification did not adversely affect system operability or availability. Additionally, the inspectors verified that the installation was consistent with applicable modification documents and conducted with adequate configuration control.

- April 21, 2004, Temporary Modification 2690709, "Implement Upgraded Temperature and Vibration Monitoring to Support Resolution of Shutdown Cooling Suction Line Vibration," Revision 1 (Unit 1)

- April 23, 2004, Temporary Modification 2691696, "Install Heat Tracing on the Unit 1 Train A Shutdown Cooling Suction Line to Reduce Vibration," Revision 0 (Unit 1)

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed portions of the announced emergency preparedness drill conducted on June 30, 2004, to evaluate emergency response organization performance by focusing on the risk-significant activities of classification, notification, and protective action recommendations. The inspectors also assessed personnel recognition of abnormal plant conditions, the transfer of emergency responsibilities between facilities, communications, and the overall implementation of the emergency plan. The drill was conducted using the Unit 1 simulator and all onsite response facilities (emergency operations facility, technical support center, and the operations support center) were activated. The scenario involved a loss of feedwater and failure of auxiliary feedwater, leading to the failure of three fission product barriers and the declaration of a general emergency.

b. Findings

No findings of significance were identified.

Cornerstone: Occupational Radiation Safety

2. RADIATION SAFETY

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspector assessed the licensee’s performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee's procedures required by Technical Specifications as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection (RP) manager, RP supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas

- Radiation exposure permit, procedure, and engineering controls and air sampler locations

- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

- Barrier integrity and performance of engineering controls of two airborne radioactivity areas

- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools

- Self-assessments, audits, licensee event reports (LERs), and special reports related to the access control program since the last inspection (no LERs or special reports identified)

- Corrective action documents related to access controls

- Radiation exposure permit briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, RP job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradient

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and RP technician performance with respect to RP work requirements

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Performance indicator events and associated documentation packages reported by the licensee in the occupational radiation safety cornerstone

The inspector completed 21 of the required 21 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspector sampled licensee submittals for the performance indicators listed below for the period from October 1, 2003, through March 31, 2004. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in Nuclear Engineering Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

The inspector completed two of the required two samples.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness Performance Indicators

Licensee records reviewed included corrective action documentation that identified occurrences in high radiation areas with dose rates greater than 1,000 millirem per hour at 30 centimeters (as defined in Technical Specification 5.7.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Engineering Institute 99-02). Additional records reviewed included as low as is reasonably achievable records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation and very high radiation areas were properly controlled.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data.

b. Findings

No findings of significance were identified.

Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors reviewed unit logs, the maintenance rule unavailability tracking database, and Technical Specification component condition records from January 2003 through March 2004 to verify the accuracy and completeness of data used to calculate and report the following performance indicators:

- Emergency ac Power System Unavailability (Units 1, 2, and 3)

- Residual Heat Removal System Unavailability (Units 1, 2, and 3)

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed a selection of CRDRs written during this period to determine if the licensee was entering conditions adverse to quality into the corrective action program at an appropriate threshold, the CRDRs were appropriately categorized and dispositioned in accordance with the licensee's procedures, and in the case of significant conditions adverse to quality, the licensee's root cause determination and extent of condition evaluation were accurate and of sufficient depth to prevent recurrence of the condition.

b. Findings

No findings of significance were identified.

2. Annual Sample Review

1. SDC Heat Exchanger Temperature Loop Channel Calibration

a. Inspection Scope

The inspectors selected CRDRs 280178 and 2686919 for detailed review. The CRDRs were associated with Technical Specification required calibration of resistance temperature detectors (RTDs) for SDC heat exchanger temperature Loops JSIA351X/Y and JSIB352X/Y. The reports were reviewed to ensure that the full extent of the issues were identified, appropriate evaluation was performed, and adequate corrective actions were identified. The inspectors evaluated the reports against the requirements of licensee Procedure 90DP-0IP10, "Condition Reporting," Revision 16, and 10 CFR Part 50, Appendix B.

b. Findings

Introduction. A Severity Level IV noncited violation finding was identified for the failure to perform a complete SDC heat exchanger temperature loop channel calibration as required by Technical Specification 3/4.3.3.5 (Improved Technical Specification requirement is 3.3.11), "Remote Shutdown System."

Description. On June 11, 1998, the licensee initiated CRDR 280178 in response to concerns regarding the implementation of surveillance requirements associated with Technical Specification 3/4.3.3.5, which specified that the channel calibration shall encompass the entire channel, including the sensor RTD. Specifically, the licensee had not performed calibrations of the SDC heat exchanger temperature instrument. The inspector determined that the licensee’s response to CRDR 280178 provided an adequate justification for why the RTDs remained functional. Although the need to calibrate the RTDs to comply with the Technical Specifications was mentioned, the licensee failed to institute corrective actions to ensure compliance with the Technical Specifications.

Due to the licensee’s failure to properly evaluate CRDR 280178 and correct the noncompliant condition, this same issue regarding Technical Specification compliance was questioned on February 27, 2004, in CRDR 2686919 following an engineering review. The inspectors determined that the licensee's response to CRDR 2686919 was adequate in that immediate actions were taken to calibrate the RTDs while the licensee determined whether a qualitative assessment had been performed as allowed by Improved Technical Specifications. Improved Technical Specifications were incorporated into the PVNGS license in August 1998, which revised the definition for channel calibration. The revised definition states, in part, that calibration of instrument channels with RTD sensors may consist of an in-place qualitative assessment of sensor behavior. The licensee determined through review of Procedure 36ST-9SI07, "Remote Shutdown Monitoring System Instrumentation Calibration for the SI System," Revision 5, that the calibration of the SDC heat exchanger temperature elements included a check of instrument output to verify that it reads as expected, which satisfies the Technical Specification required in-place qualitative assessment.

The inspectors identified that the licensee failed to properly review this condition for reportability during their evaluation of CRDR 2686919. Procedure 36ST-9SI07 provided for an in-place qualitative assessment when revised on February 17, 2000. The licensee based the reportability review on the current surveillance procedure revision and incorrectly concluded that the qualitative assessment had been performed since the implementation of improved Technical Specifications in August 1998. The instrument output verification was incorporated into Procedure 36ST-9SI07 on February 17, 2000. The inspectors determined that the Technical Specification required in-place qualitative assessment for the SDC heat exchanger temperature instruments had been performed during channel calibrations since February 2000. Nevertheless, with respect to the SDC heat exchanger temperature instruments, the inspectors determined that between August 1998 and January 25, 2001 (Unit 1), and on June 15 (Unit 2) and May 18, 2001 (Unit 3), the licensee did not perform either a qualitative assessment of sensor behavior or a calibration of the sensor. The inspectors identified that the past noncompliant condition was reportable per 10 CFR 50.73, "Licensee Event Report System."

Analysis. This finding is greater than minor because the licensee's failure to report the condition impacted the NRC's ability to perform its regulatory function. This finding is associated with the mitigating systems cornerstone. This finding was considered applicable to traditional enforcement. Although the significance determination process is not designed to assess significance of violations that potentially impact or impede the regulatory process, the finding can be assessed using the significance determination process. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," this finding is determined to be of very low safety significance because it only affected the mitigating system cornerstone and the RTDs were found to be within calibration.

Enforcement. Technical Specification 3.3.11, "Remote Shutdown System," requires that the remote shutdown system instrumentation functions in Table 3.3.11-1 be operable.

Item 4.a of Table 1 includes SDC heat exchanger temperature. Technical Specification Surveillance Requirement 3.3.11.3 required that a channel calibration be performed every 18 months. Technical Specification Surveillance Requirement 3.0.1 specified that a failure to meet a surveillance requirement is a failure to meet the Technical Specification. Contrary to this, the licensee failed to complete channel calibrations on the SDC heat exchanger temperature elements. Specifically, the licensee did not test the RTD or perform an in-place qualitative assessment. The inspectors also determined that the licensee's failure to implement effective corrective actions following the identification of the issue documented in CRDR 280178 resulted in the violation of Technical Specification 3.3.11 existing for an extended duration. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CRDR 280178. NCV 05000528;05000529;05000530/2004003-06, "Failure to Perform a Complete SDC Heat Exchanger Temperature Loop Channel Calibration."

2. Failure to Perform Monthly Reviews to Ensure Excessive Hours Have Not Been Assigned

a. Inspection Scope

The inspectors selected CRDR 2646228 for detailed review. The CRDR was written to describe a contract employee that exceeded the 72 hours in 7 days limit due to the change from 10-12 hours shifts. The report was reviewed to ensure that the full extent of the issue was identified, appropriate evaluation was performed, and adequate corrective actions were identified. The inspectors evaluated the reports against the requirements of licensee Procedure 90DP-0IP10, "Condition Reporting," Revision 16, and 10 CFR Part 50, Appendix B. The inspectors also reviewed overtime limitation exception reports generated during the latest refueling outages for Units 1 and 3.

b. Findings

Introduction. The inspectors identified a noncited violation of Technical Specification 5.2.2.d for the failure of authorized individuals to review monthly overtime to ensure that excessive hours have not been assigned.

Description. Technical Specification 5.2.2.d, states, in part, that controls shall be included in procedures such that individual overtime shall be reviewed monthly by authorized individuals or designees to ensure that excessive hours have not been assigned. Procedure 01DP09EM01, step 4.5, states, in part, that the excess hours report will be distributed to the applicable individuals that are one level higher than a department leader of the applicable departments that this control applies to. This report shall be reviewed monthly to ensure excessive hours have not been assigned and the overtime limitations have not been violated. The inspectors identified, based on interviews with the maintenance director, that he had not been performing these reviews because the excess hours report had not been distributed to site maintenance management since an electronic reporting system was instituted in 2001.

Analysis. The finding is greater than minor because if left uncorrected it could become a more significant safety concern, in that exceeding the NRC Generic Letter 82-02, "Nuclear Power Plant Staff Working Hours," guidelines for overtime limits can be a contributor to worker fatigue. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," this finding is determined to be of very low safety significance because there were no known actual adverse plant or equipment conditions that could be attributed to worker fatigue.

Enforcement. Technical Specification 5.2.2.d, states, in part, that controls shall be included in procedures such that individual overtime shall be reviewed monthly by authorized individuals or designees to ensure that excessive hours have not been assigned. Contrary to the above, since 2001, the licensee has failed to perform the monthly reviews to ensure excessive hours have not been assigned. Because the finding is of very low safety significance and has been entered into the corrective action program as CRDR 2727646, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528; 05000529; 05000530/2004003-07, "Failure to Perform Monthly Reviews to Ensure Excessive Hours Have Not Been Assigned."

3. Semiannual Review

a. Inspection Scope

During the inspection period, the inspectors reviewed the licensee’s corrective action program, interviewed licensee personnel, and reviewed the maintenance rule program documents to identify any significant adverse system or equipment trends. The inspector performed a detailed review of the following problems that consisted of multiple examples:

- Fourteen examples where inappropriate lubricants were utilized on plant equipment. The inspector verified that all examples were minor and did not have an adverse impact on plant components.

- Eight examples of motor-operated valve problems. In three cases, valve thrust was found in excess of administrative design limits. In one case, valve thrust was below design limits. In another example a valve failed to close on demand. The final four problems involved high running loads, the inability to place an actuator in the manual mode, an out of alignment motor pinion gear key, and a broken worm shaft clutch broken dog. The inspectors verified that none of the problems were the result of a programmatic breakdown and that all issues that affected operability were properly resolved.

b. Findings

No findings of significance were identified.

4. Section 2OS2 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding access controls to radiologically significant areas and radiation worker practices. No findings of significance were identified.

4OA3 <u>Event Followup (71153, 71111.14)</u>

1. <u>Loss of Offsite Power, Subsequent Unit Reactor Trips, and Unit 2 Emergency Diesel Generator Train A Failure to Maintain Voltage</u>

The inspectors evaluated plant conditions, equipment performance, and licensee actions related to the LOOP and reactor trips on Units 1, 2, and 3. Units 1 and 3 declared a notice of unusual event for this LOOP and Unit 2 declared an alert due to emergency diesel generator Train A not maintaining voltage. An NRC Augmented Inspection Team was chartered to review the complete LOOP. The results will be documented in NRC Augmented Inspection Report 05000528; 05000529; 0500530/2004012.

2. <u>Failure to Have Written Instructions for Testing a Remotely Controlled Submersible Vehicle in the Unit 1 SFP</u>

a. <u>Inspection Scope</u>

The inspectors evaluated plant conditions, equipment performance, and licensee actions related to a 3EF increase in the Unit 1 SFP due to operations isolating one of the SFP cooling pumps to allow the retrieval of a remotely controlled submersible vehicle which had entered into the common suction of the SFP cooling system.

b. <u>Findings</u>

<u>Introduction</u>. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for the failure of the licensee to have written instructions for testing a remotely controlled submersible vehicle in the Unit 1 SFP. The submersible unexpectedly entered into the common suction for the pool cooling system. At the time of the event, the unit was in refueling operations with 164 of the 241 fuel assemblies unloaded into the SFP.

<u>Description</u>. On April 12, 2004, during refueling operations in Unit 1, an RP technician was testing, without written instructions and/or operations permission, a remotely controlled submersible vehicle in the SFP in preparation for a scheduled inspection of the reactor vessel the following day. While the technician discussed the process for removing the vehicle from the SFP with the fuel building RP technician, the vehicle began to sink and entered into the SFP cooling pump combined suction. The vehicle was prevented from continuing through the pipe by the RP technician and spent fuel

handling machine operator holding the tether until one of the two operating SFP cooling pumps could be stopped by operations. With only one SFP cooling pump operating, the vehicle was pulled from the pipe and retrieved from the pool. A temperature increase of 3°F (97E to 100°F) was noted by operations during the 16 minutes that the Train A SFP cooling pump was secured. The SFP cooling pump was subsequently restarted and core offload resumed. At the time of the event, the plant was in refueling operations with 164 of the 241 fuel assemblies unloaded into the SFP. The licensee reported that the time-to-boil in the SFP was 6 hours.

Analysis. The finding is greater than minor because it affected the configuration control and human performance attributes of the initiating events cornerstone objective. 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 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.

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality 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, the licensee failed to have written instructions to test a remotely controlled submersible vehicle in the Unit 1 SFP. Because this violation is of very low safety significance and has been entered into the corrective action program as CRDR 2697384, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2004003-08, "Failure to Have Instructions for Testing a Submersible in the Unit 1 SFP."

4OA4 Crosscutting Aspects of Findings

1. Cross-References to Human Performance Findings Documented Elsewhere

Section 1R19 describes a finding where inattention to detail resulted in the inadequate removal of isopropyl alcohol from an MFW pump Train A turbine casing and caused a flash fire.

Section 1R20 describes a finding for poor decision making and questioning attitude that resulted in failing to correct a degraded refueling machine equipment condition.

Section 1R22 describes a finding where poor operator decision making, questioning attitude, awareness of plant conditions, and communications between operators resulted in a loss of letdown and pressurizer level transient.

Section 4OA3 describes a finding associated with poor decision making and lack of questioning attitude from an RP technician which lead to a remotely controlled submersible being ingested into the common suction for the Unit 1 SFP cooling system during refueling operations.

Section 4OA5 of this report and Section 4OA2 of NRC Inspection Report 05000528; 05000529; 05000530/2003004 described a finding that involved mispositioned valves, improper procedure implementation, and a lack of operator awareness of plant conditions that resulted in SFP inventory loss events.

4OA5 Other Activities

1. (Closed) Unresolved Item (URI) 05000530/2003004-03: Nuclear Assurance Department's Concurrence for Significant CRDR 2599869

Introduction. A Green noncited violation was identified for the failure to identify the root cause of SFP inventory loss events and implement corrective actions to preclude recurrence.

Description. NRC Inspection Report 05000528;05000529;05000530/2003004 documents inspector observations associated with the review of two significant investigations, CRDRs 2599869 and 2451670, performed by the licensee to correct a negative trend in SFP inventory control. Deficiencies involving inadequate root cause identification and untimely corrective action implementation were identified by the inspector during the review. The Nuclear Assurance Department reviewed CRDR 2599869 and concluded that the cause evaluation was inadequate and rejected the significant investigation. As a result of the untimely significant investigations and corrective action implementation, two additional SFP inventory loss events occurred in March and April 2004. Similar to the events identified in CRDR 2599869, causes of the 2004 events included improper positioning of fuel pool cleanup suction isolation Valve XPPCNV004 and inadequate level monitoring, both locally and in the control room. Significant Investigation Report 2599869 was approved April 22, 2004, and adequately identified the root causes of past SFP inventory events and corrective actions to preclude repetition. The root causes identified included: (1) the combination of the inaccuracy of the position indicator for gear-driven Valve Operators 2PPCNV004 and 3PPCNV004, and the overconfidence and complacency of the auxiliary operators, lead the auxiliary operator to believe that the valve was closed on initial positioning; the resultant valve not in the full closed position caused the change in SFP level; and (2) the expectation that local monitoring of the SFP level could be maintained a top priority, given the embedded distraction inherent in performing tasks in the pool cooling system, was not realistic or consistently achievable. Remote detection and warning of an SFP level decrease was not provided for prompt event mitigation.

Analysis. The finding is greater than minor because it affected the configuration control and human performance attributes of the initiating events cornerstone objective. 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 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.

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 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 actions taken to preclude repetition. Contrary to the above, the licensee failed to identify the root cause of SFP inventory loss events and did not implement corrective actions to preclude repetition in a timely manner. Because this violation is of very low safety significance and all SFP draindown events have been entered in the corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528; 05000529; 05000530/2004003-09, "Failure to Prevent Loss of SFP Inventory Events Through Timely Corrective Actions."

2. (Closed) URI 05000529/2003009-02: Failure to Follow Heavy Load Movement Procedure

Introduction. NRC Inspection Report 05000529/2003009 described a noncited violation for failing to follow Procedure 31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revision 8, preceding a heavy load drop inside containment. The finding was identified as a URI pending the completion of a significance determination. The result of the determination was that the finding was of very low safety significance.

Description. During the inspection documented in NRC Inspection Report 05000529/2003009, the inspectors identified a noncited violation having potential safety significance greater than very low significance, involving the 12-24 inch drop of a 7000-pound steam generator snubber lever plate during the steam generator replacement outage. The inspectors had determined that the licensee failed to follow it's control of heavy loads procedure which led to the nonuse of numerous barriers which should have been in place to prevent the load drop. The significance of this finding had not been determined at the conclusion of the inspection.

Analysis. The finding was greater than minor because it affects the equipment performance and human performance attributes of the initiating event cornerstone objective to limit the likelihood of events that challenge safety functions during shutdown conditions. During the current inspection period, the inspectors and senior reactor analyst determined that the issue was of very low safety significance (Green). The following assumptions were used during the Manual Chapter 0609, "Significance Determination Process," Appendix G, "Shutdown Operations Significance Determination Process," evaluation:

- The closest RCS piping in the horizontal direction from the lever plate was the RCS Loop 1A cold leg. The licensee performed a calculation which concluded that, if the lever dropped on the cold leg, it would have remained intact.

- The pressurizer surge line was approximately 6.5 feet horizontally and 30 feet vertically from the lever plate and was not a realistic target for the lever plate. However, even if the pressurizer surge line had been severed, residual heat removal operation would not have been impacted because the surge line exited from the top of the RCS hot leg.

- There were no other targets of RCS piping or interconnected systems in the postulated drop zone of the lever plate that could have impacted residual heat removal operation or that would have impaired equipment that could mitigate a loss of RCS inventory (such as high pressure injection or containment spray).

- Containment closure could have been achieved, if necessary, in less than 25 minutes of a loss of RCS inventory, and containment closure would not have been impacted by a postulated break in the pressurizer surge line.

Based on the above, the senior reactor analyst concluded that this finding did not significantly increase the likelihood of losing the RHR function and did not significantly increase the likelihood that systems that could mitigate a loss of RHR function would be degraded. Therefore, this finding was of very low safety significance.

<u>Enforcement</u>: NRC Inspection Report 05000529/2003009 described a noncited violation for failing to follow Procedure 31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revision 8, preceding a heavy load drop inside containment. Because the finding is of very low safety significance and has been entered into the corrective action program as CRDR 2639721, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2004003-10, "Failure to Follow Heavy Load Movement Procedure."

3. <u>(Closed) URI 05000529/2003005-03</u>: Missing Bolts on Support for Main Steam Line Whip Restraint

<u>Introduction</u>. A Green self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified for the failure to secure a main steam line pipe whip restraint inside the Unit 2 containment in accordance with design drawings.

<u>Description</u>. As described in NRC Inspection Report 050000528; 05000529; 05000530/2003005, the inspectors identified an apparent violation of NRC requirements

(with significance to be determined) for a finding involving the discovery that Unit 2 Steam Generator 2 Pipe Whip Restraint Hanger 02-SG-042-H-890 was missing fasteners required by Drawing 13-C-ZCS-541. The inspectors determined that this apparent violation was a URI pending determination of its risk significance. A concern existed that with the pipe whip restraint fasteners absent, the containment liner or the containment spray headers could be damaged by pipe whip following a main steam line break accident. The licensee initiated an analysis to evaluate the consequences of a postulated pipe whip in this condition.

Analysis. The finding is greater than minor since it is associated with the equipment performance attribute of the barrier integrity cornerstone and affects the cornerstone objective of protecting the containment barrier from radionuclide releases caused by accidents or events. During the current inspection period, the inspectors and a senior reactor analyst determined that the issue was of very low safety significance and did not require a Phase 2 analysis using Manual Chapter 0609, "Significance Determination Process," Appendix H, "Containment Integrity SDP." The factors causing the issue to be of very low safety significance were:

- The finding did not represent an actual open pathway in the physical integrity of reactor containment.
- The finding did not represent an actual reduction of the atmospheric pressure control function of the reactor containment.
- The licensee’s completed analysis, 0001-04-KB-APS-13-CC-ZC-0165A, "Main Steam Line (Unit 2) SG#2 Pipe Whip Restraints," Revision 0, concluded that the support structure and the pipe whip restraint would likely restrain the pipe lines upon the occurrence of a steam line break without the required fasteners. Neither the containment liner nor the containment spray headers would be impacted.

Enforcement. Because this failure to comply with 10 CFR Part 50, Appendix B, Criterion V, is of very low safety significance and has been entered into the corrective action program as CRDR 2643347, this violation is being treated as an noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2004003-11, "Missing Bolts on Support for Main Steam Line Whip Restraint."

4. (Closed) LER 05000528; 05000529; 05000530/2003003-00, Technical Specification Violation for Failure to Meet SDC Trains OPERABLE Action Statements

A licensee-identified noncited violation of Technical Specification 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," was identified for a temporary, nonseismically qualified containment pedestal crane lifting heavy loads over the operating train of SDC in all three units during SDC operations, thereby rendering the operating train of SDC inoperable. On September 19, 2003, during preparations for the Unit 2 steam generator replacement project, the licensee identified that the temporary containment pedestal crane was not seismically qualified. The containment pedestal crane was installed in the containment building at the beginning of each refueling outage after the unit entered cold shutdown. It was removed prior to exiting cold shutdown. The containment pedestal crane had been used for rigging maintenance equipment needed during refueling outages for all three units since May 1995. The licensee determined that the crane was not analyzed for operation with a loaded hook during a seismic event. If a heavy load lift was in progress using the containment pedestal crane and a seismic event occurred while the load was above equipment needed for SDC when fuel was in the reactor vessel, and the load was dropped, a potential existed for SDC capability or RCS inventory to be lost. This constituted a violation of Technical Specifications 3.4.7, 3.9.4, and 3.9.5, which required operability of SDC systems during shutdown conditions. Upon discovery of this finding, the licensee restricted use of the containment pedestal crane. The maximum load lifted with the containment pedestal crane was approximately 3500 pounds.

The finding is greater than minor because it affects the mitigating systems cornerstone objective and because the finding could be reasonably viewed as a precursor to a significant event. In determining the significance of this finding, the inspectors assumed the possibility of a seismic event resulting in the loss of a single train of SDC. All mitigating systems remained available to the operators and the finding did not increase the likelihood of a fire or flooding. The inspectors, in coordination with the senior reactor analyst, determined that the finding was of very low safety significance based on the number of outages the containment pedestal crane was used, an estimate of the number of lifts and load paths over reactor coolant system targets, a qualitative review of the fragility of the containment pedestal crane, and the low seismicity of the area. This licensee-identified finding involved a violation of Technical Specification 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level." The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

5. <u>Reactor Pressure Vessel (RPV) Head and Vessel Head Penetration Nozzles Temporary Instruction (TI) 2515/150</u>

In October 2002, the inspectors completed the review of the licensee’s Unit 1 RPV head bare metal visual examination using TI 2515/145. This review was documented in NRC Inspection Report 0500528/2002006.   Per TI 2515/150, Section 07, "Expiration," the October 2002 completion of TI 2515/145 was credited as one of the two required TI 2515/150 reviews. The second occurrence of TI 2515/150 for Unit 1 is documented below. Therefore, TI 2515/145 is closed for Unit 1.

The first occurrence of TI 2515/150 was documented for Unit 2 in NRC Inspection Report 0500529/2003-005. The first occurrence of TI 2515/150 was documented for Unit 3 in NRC Inspection Report 0500530/2003003.

Susceptibility Ranking Calculation

a. Inspection Scope

On April 5-23, 2004, the inspectors performed NRC Inspection Manual TI 2515/150 for Unit 1 during Cycle 11 Refueling Outage 1R11. The inspectors reviewed the licensee's inspection plan in response to NRC Order EA-03-009, which established interim inspection requirements for RPV heads.

The inspectors reviewed the susceptibility ranking calculation to verify that appropriate plant-specific information was used as input. The calculation determines the effective degradation years, which is the effective full power years, normalized to 600EF. Two periods were used to determine RPV head temperature and corresponded to the periods before and after implementation of T-hot reduction, which reduced T-hot from 621EF to approximately 612EF to minimize steam generator tube degradation. The head temperature for each period was based on using a combination of an evaluation to calculate fluid temperature in the upper head based on mixing of bypass flow through different paths and heated junction thermocouple data. The more conservative of the two temperatures was used for each period.

The inspectors noted that Unit 1 was projected to be in the highly susceptible category at the end of Cycle 11. Required inspections for the refueling outage were bare metal visual examination of 100 percent of the RPV head surface (Order, Section IV.C.(1)(a)), ultrasonic testing of each RPV head penetration nozzle from 2 inches above the J-groove weld to the bottom of the nozzle (Order, Section IV.C.(1)(b)(i)), or eddy current testing of the wetted surface of each J-groove weld and RPV head penetration nozzle base material to at least 2 inches above the J-groove weld (Order Section IV.C.(1)(b)(ii)). Because of hardships the licensee had with the ability to perform required inspections, two relaxation requests submitted to the NRC were approved based on the demonstration of good cause for the proposed relaxations. The first proposed alternative examination was to perform a bare metal visual examination of the one RPV head vent line nozzle in accordance with Order Section IV.C.(1)(a), since internal volumetric or surface examination would be difficult and would require the removal of the welded orifice and testing of the remaining control element drive mechanism nozzles per Order Section IV.C.(1)(b). The second proposed alternative examination was to perform ultrasonic testing of each nozzle from 2 inches above the J-groove weld to approximately 0.4" above the top of the nozzle’s chamfer face control element drive mechanism since ultrasonic scans in the area below 0.4" to the bottom of the nozzle do not yield useful data because of the geometry of the nozzle and funnel.

b. Findings

No findings of significance were identified.

Volumetric and Surface Examinations

a. Inspection Scope

The inspectors verified that the licensee's volumetric inspection plan and critical performance objectives were incorporated into site procedures. They also interviewed plant inspection personnel and contractors performing the inspections to determine their understanding of inspection standards and acceptance criteria required during data gathering and analysis. The inspectors reviewed the Westinghouse field service procedures, which governed the instrument calibration, data gathering, and data analysis requirements for ultrasonic and eddy current testing. Nuclear Reactor Regulation personnel, in conjunction with the inspectors, reviewed the qualification of these methods and their ability to determine flaws in J-groove welds and base metals associated with primary water stress corrosion cracking. The inspectors reviewed licensee and contractor qualifications and certification records which were obtained through a combination of written and practical examinations. The inspectors conducted interviews with plant engineers and Westinghouse contractors to determine their training, background, basis used for certifications, and expertise in conducting and analyzing these examinations. The inspectors also observed equipment operation during data gathering for 10 nozzles and data analysis for 100 percent of the head penetration nozzles. The inspectors compared 4 nozzles of special interest to data collected from the previous outage and determined that there were no changes in the anomalies.

b. Findings

No findings of significance were identified.

Bare Metal Visual Examinations

a. Inspection Scope

The inspectors observed the video acquired during visual inspection of the RPV head vent line nozzle and noted that the camera and remote monitoring equipment used during the examination process provided adequate visual clarity. The inspectors reviewed certification records and discussed the qualifications and experience of the examiners. The inspectors verified that a clear 360E observation of the nozzles was completed and that no evidence of cracking or boric acid crystals were present. There were no boron deposits, debris, or insulating material which masked the ability to identify the existence of boric acid. There were no structural interferences which impeded the ability to complete the bare metal visual inspections. The inspectors determined that the licensee had procedures in place to identify leakage from pressure retaining components located above the RPV head.

b. Findings

No findings of significance were identified.

6. RPV Lower Head Penetration Nozzles (TI 2515/152)

a. Inspection Scope

On April 8-23, 2004, the inspectors reviewed the licensee's response to NRC Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity." The response described the licensee's commitment to perform a bare metal visual inspection of all 62 nozzle penetrations in the lower reactor head of all three units. The inspectors reviewed the licensee's procedures for the inspection of the Unit 1 lower head penetrations. The inspector also reviewed the qualification and certifications for the personnel performing the inspections.

The inspectors reviewed a video tape of all nozzle inspections. The inspections covered a full 360E of all 62 nozzle penetrations. The camera and remote monitoring equipment used during the examination process provided adequate visual clarity. The inspectors verified that a clear 360E observation of the nozzles was completed and that no evidence of cracking or boric acid crystals were present. The inspectors determined that there was no debris, insulation, or boric acid deposit on the RPV lower head. TI 2515/152 has been completed on Units 1 and 2.

b. Findings

No findings of significance were identified.

7. Reactor Containment Sump Blockage - NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors" (TI 2515/153)

Generic Safety Issue 191 was established to determine whether the transport and accumulation of debris in pressurized water reactor containments following a loss of coolant accident (or other high energy line break, if recirculation is credited) will impede the long-term operation of the ECCS or containment spray system. In the event of a loss of coolant accident, materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, would be damaged and dislodged.

The inspector reviewed the licensee's response and supporting basis which showed that the ECCS and containment spray system recirculation functions have been analyzed with respect to the potentially adverse postaccident debris blockage effects as specified in the bulletin. The inspector assessed that this determination is based on a mechanistic (plant-specific) evaluation of debris generation, transport, and accumulation rather than arbitrary (generic) assumptions.

The inspector also confirmed that the licensee performed walkdowns of their containments to quantify potential debris sources and check for gaps in the sumps' screened flowpath and for major obstructions in containment upstream of the sumps. The inspector also assessed any sump-related modifications.

TI 2515/153 has been completed for Units 1 and 2.

Interim Compensatory Measures

a. Inspection Scope

Possible interim compensatory measures may include, but are not limited to, the following:

- Operator training on indications of and responses to sump clogging
- Procedural modifications, if appropriate, that would delay the switchover to containment sump recirculation (e.g., shutting down redundant pumps that are not necessary to provide required flows to cool the containment and reactor core and operating the containment spray system intermittently)
- Ensuring that alternative water sources are available to refill the refueling water storage tank or to otherwise provide inventory to inject into the reactor core and spray into the containment atmosphere
- More aggressive containment cleaning and increased foreign material controls
- Ensuring containment drainage paths are unblocked
- Ensuring sump screens are free of adverse gaps and breaches

b. Findings

The licensee informed operators of the actions for potential blockage but decided to defer any procedure changes until the Combustion Engineering Owners Group evaluates the bulletin.

No findings of significance were identified.

Debris Sources in Containment

a. Inspection Scope

The inspectors reviewed the potential debris sources in containment described in Updated Final Safety Analysis Report, Section 6.2.2, and in Calculation 13-MC-SI-309, "Containment Sump Blockage," Revision 3. The inspectors also toured the Unit 1 containment to determine that there were no additional debris sources.

b. Findings

The inspectors toured containment and identified exposed Fiberfrax due to improper insulation installation in five bio-shield wall penetrations. This condition was evaluated by the licensee in CRDR 2710401. The engineering evaluation determined that the quantity of Fiberfrax identified would not impact ECCS sump operability since the design basis accident scenario bounds all fiber and pads located in the five penetrations for ECCS sump clogging concerns.

No findings of significance were identified.

Containment Sump Inspection and Design

a. Inspection Scope

The inspectors reviewed the design of the containment sumps which are designed to be reservoirs of water to the ECCS following a loss of coolant accident. The inspectors verified that the sump configuration satisfied design requirements in that particles greater than 3/16-inch diameter were precluded from entering the ECCS sump.

b. Findings

No findings of significance were identified.

8. Offsite Power System Operational Readiness TI 2515/156

a. Inspection Scope

The inspectors collected data from licensee maintenance records, event reports, and corrective action documents and procedures and through interviews of station engineering, maintenance, and operations staff as required by TI 2515/156. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, "General Design Criterion (GDC) 17"; Criterion XVI of Appendix B to10 CFR Part 50; Plant Technical Specifications for offsite power systems; 10 CFR 50.63; 10 CFR 50.65(a)(4); and licensee procedures. Documents reviewed for this TI are listed in the attachment.

b. Findings

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to the headquarters staff for further analysis. This completes TI 2515/156 for the Palo Verde Nuclear Generating Station.

4OA6 Meetings, Including Exit

On April 22, 2004, the senior reactor inspector presented the inservice inspection results to Mr. G. Overbeck, Senior Vice President, Nuclear, and other members of his staff. The licensee acknowledged the findings.

On April 23, 2004, the health physicist inspector presented the radiation safety inspection results to Mr. G. Overbeck, Senior Vice President, Nuclear and other members of his staff. The licensee acknowledged the findings.

On May 21, 2004, the resident inspectors presented partial integrated inspection results to Mr. G. Overbeck, Senior Vice President, Nuclear, and other members of his staff. The licensee acknowledged the findings.

On June 1, 2004, a subsequent conference call was held with Mr. Tom Weber, Section Leader, and licensing personnel to discuss the final conclusion and characterization of the findings for the inservice inspection. The licensee acknowledged the findings.

On July 8, 2004, the resident inspectors presented the integrated inspection results to Mr. G. Overbeck, Senior Vice President, Nuclear, and other members of his staff. The licensee acknowledged the findings.

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

40A7 Licensee-identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-600, for being dispositioned as an NCV.

- 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 3a, requires procedures containing instructions for operation of the RCS. Contrary to the above, on May 5, 2004, the licensee did not follow Procedure 40ST-RC01, "RCS and Pressurizer Heatup and Cooldown," Revision 13, step 8.1.2, while preparing for Mode 4 entry. Step 8.1.2 provided instructions for the appropriate instruments to be used during RCS heatup and cooldown. Operators were inappropriately controlling RCS temperature with SDC heat exchanger Train A outlet temperature on the emergency response facility data acquisition and data system to maintain RCS temperature in Mode 5, just below Mode 4 conditions. Due to an unknown emergency response facility data acquisition and data system deficiency, the indication used by operators to control temperature indicated approximately 17EF lower than the procedurally approved temperature indications. Operators identified the condition when they unexpectedly noted, on trend Recorder SIATT351Y, that RCS temperature was at the upper limit for Mode 5. An RCS cooldown was initiated to establish more temperature margin from Mode 5 conditions. This was identified in the licensee's corrective action program as CRDR 2706235. This finding is of very low safety significance because the condition was identified prior to making an inadvertent mode change.

- Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the applicable procedures referenced in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Appendix A, Section 7, requires procedures for access control to radiation areas including a radiation work permit system. On April 8, 2004, a contract worker entered the Unit 1 containment building on Radiation Exposure Permit 1-305B, Task 1. The radiation exposure permit required the worker to review current radiological survey data and receive an area specific RP prejob briefing prior to entry; however, the worker failed to comply with these requirements prior to entering the work area. This event was described in the licensee's corrective action program as CRDR 2696454. The finding was determined to be of very low safety significance because the violation did not involve as low as reasonably achievable planning or work controls, no individual received an overexposure or a substantial potential for overexposure, and the ability to assess dose was not compromised.

- Technical Specification 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," requires one SDC loop to be operable and in operation. Contrary to the above, during refueling outages in all three units dating back to May 1995, the licensee used a temporary, nonseismically qualified containment pedestal crane to lift heavy loads over the operating train of SDC which rendered the operating train of SDC inoperable. This event was described in the licensee's corrective action program as CRDR 2636484. In determining the significance of this finding, the inspectors assumed the possibility of a seismic event resulting in the loss of a single train of SDC. All mitigating systems remained available to the operators and the finding did not increase the likelihood of a fire or flooding. The inspectors, in coordination with the senior reactor analyst, determined the finding was of very low safety significance based on the number of outages the containment pedestal crane was used, an estimate of the number of lifts and load paths over RCS targets, a qualitative review of the fragility of the containment pedestal crane, and the low seismicity of the area.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Brandjes, Department Leader, Maintenance
D. Carnes, Director, Nuclear Assurance and Regulatory Affairs
W. Chapin, Department Leader, Nuclear Fuels Management
C. Churchman, Director, Steam Generator Replacement Project
M. Fladager, Department Leader, Radiation Protection
J. Gaffney, Director, Radiation Protection
T. Gray, Department Leader, Radiation Protection
M. Grigsby, Unit Department Leader, Operations
D. Hautala, Senior Engineer, Regulatory Affairs
R. Henry, Site Representative, El Paso Gas and Electric
D. Kanitz, Senior Engineer, Regulatory Affairs
P. Kirker, Unit Department Leader, Operations
D. Mauldin, Vice President, Engineering and Support
M. McGhee, Unit Department Leader, Operations
M. Milton, Section Leader, Inservice Inspection
G. Overbeck, Senior Vice President, Nuclear Operations
S. Peace, Consultant, Owners Services
R. Pontes, Section Leader, Steam Generator Replacement Project
M. Powell, Department Leader, Maintenance Engineering
J. Taylor, Department Leader, Operations
C. Seaman, Director, Nuclear Fuels Management
D. Smith, Plant Manager, Nuclear Production
E. Sterling, Section Leader, Nuclear Assurance
K. Sweeney, Section Leader, Steam Generator Project Group
T. Weber, Section Leader, Regulatory Affairs
M. Winsor, Director, Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000530/2004003-01	NCV	SFP Water Siphon due to Check Valve Failure (Section 1R14)
05000528;05000529; 05000530/2004003-02	NCV	Containment Purge Penetration Nonconformance (Section 1R15)

05000528/2004003-03	NCV	Fire that Occurred During Welding Activities on the MFW Pump Turbine Train A (Section 1R19)
05000528/2004003-04	NCV	Core Alterations with Degraded Refueling Machine (Section 1R20)
05000528/2004003-05	FIN	Pressurizer Level Transient Above Technical Specification Limits (Section 1R22)
05000528;05000529; 05000530/2004003-06	NCV	Failure to Perform a Complete SDC Heat Exchanger Temperature Loop Channel Calibration (Section 4OA2)
05000528;05000529; 05000530/2004003-07	NCV	Failure to Perform Monthly reviews to Ensure Excess Hours Have Not Been Assigned (Section 4OA2)
05000528/2004003-08	NCV	Failure to Have Instructions for Testing a Submersible in the Unit 1 SFP (Section 4OA3)
05000528;05000529; 05000530/2004003-09	NCV	Failure to Prevent Loss of SFP Inventory Events Through Timely Corrective Actions (Section 4OA5)
05000529/2004003-10	NCV	Failure to Follow Heavy Load Movement Procedure (Section 4OA5)
05000529/2004003-11	NCV	Missing Bolts on Support for Main Steam Line Whip Restraint (Section 4OA5)

Closed

05000530/2003004-03	URI	NAD's Concurrence for Significant CRDR 2599869 (Section 4OA5)
05000529/2003009-02	URI	Failure to Follow Heavy Load Movement Procedure (Section 4OA5)
05000529/2003005-03	URI	Missing Bolts on Support for Main Steam Line Whip Restraint (Section 4OA5)
05000528/2003003-00; 05000529/2003003-00; 05000530/2003003-00;	LER	Technical Specification Violation for Failure to Meet SDC Trains OPERABLE Action Statements (Section 4OA5)

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 1R04: Equipment Alignment

Procedure

40OP-9DG02, "Emergency Diesel Generator B," Revision 31

Plant Drawings

01-M-SIP-001, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 25
01-M-SIP-002, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 24
01-M-DGP-001, "P&I Diagram - Diesel Generator Fuel Oil and Transfer System," Revision 11
01-M-DGP-001, "P&I Diagram - Diesel Generator System, Sheet 1," Revision 44
01-M-DGP-001, "P&I Diagram - Lube Oil, Diesel Generator System, Sheet 3," Revision 44
01-M-DGP-001, "P&I Diagram - Jacket Water, Diesel Generator System, Sheet 4," Revision 44
01-M-DGP-001, "P&I Diagram - Fuel Oil, Diesel Generator System, Sheet 7," Revision 44
01-M-DGP-001, "P&I Diagram - Diesel Generator System, Sheet 9," Revision 44

Section 1R07: Heat Sink Performance

CRDR

2653867

Procedures

73DP-9ZZ10, "Guidelines for Heat Exchanger Thermal Performance Analysis," Revision 4

70TI-9EW01, "Thermal Performance Testing of Essential Cooling Water Heat Exchangers," Revision 4

WO

WO 2628693

Nondestructive Examination Activities Reviewed

System/Line No/Compoent ID	Weld Number	Exam Method
SG/SG-E-70-DLBB-12	51-30	UT/MT
SG/SG-E-70-DLBB-12	51-31	UT/MT
SG/SG-E-70-DLBB-12	51-32	UT/MT
SG/SG-E-70-DLBB-12	51-33	UT/MT
SG-E-207-DLLBB-28	48-2	MT
SG-E-207-DLLBB-28	48-3	MT

SG-E-207-DLLBB-28	48-4	MT
13SG-070-H-006	Support	VT-3
SI-A-307-GCBC-24	84-26	UT/PT
IJSIBFE0348	250635-1	RT

Examinations from Previous Outage with Recordable Indications

Weld 74-37 (Determined to be geometry) (SI system)
Weld 59-16 (Determined to be geometry) (SG Downcomer)
Weld 84-26 (Determined to be geometry) (SI system)
Weld 74-37 (Determined to be geometry) (SI system)
Weld 2581743-2 (Feedwater) RF11
Weld 2581743-3 (Feedwater) RF11
Weld 2581743-6 (Feedwater) RF11

ASME Code Repair and Replacement

WO 2590635; Flange to Pipe; Containment spray discharge line; Weld 250635-1

Miscellaneous Documents

Unit 1 Summary Reports for RF8, 9, and 10

Relief Requests, submitted dated April 10, 2000

Deviations (Technical Justification) from NEI 97-06 (99-SGPG-TJ-002, -003, -004, -005, -007; and 2001-SGPG-TJ-011)

Relief Request 26 (Supplement 10 to Appendix VIII)

Procedures

73TI-9ZZ07, "Liquid Penetrant Examination," Revision 9

73TI-9ZZ10, "Ultrasonic Testing Examination Of Welds In Ferritic Components," Revision 10

73TI-9ZZ17, "Visual Examination Of Welds, Bolting, And Components," Revision 8

73TI-9ZZ79, "ASME Section XI Appendix VIII Ultrasonic Testing Of Ferritic Piping," Revision 3

73TI-9ZZ80, "ASME Section XI Appendix VIII Ultrasonic Testing Of Austenitic Piping," Revision 3

73TI-9RC01, "Steam Generator Eddy Current Examinations," Revision 23

81CP-9RC28, "Checkout and Operation of the Steam Generator Tube In Situ Pressure Test System 2"

81DP-9RC01, "PVNGS Steam Generator Degradation Management Program," Revision 3

81CP-9RC29, "In-Situ Pressure Test Using the Computerized Data Acquisition System (Westinghouse PVNGS-010, Revision 5)," Revision 4

81CP-9RC19, "Welded Tube/Tubesheet Plug Removal utilizing the Phase III Drill Assembly," Revision 3

CRDRs

658882, 2682277, 2670186, 2668058, 2667731, 2661027, and 2699053

Section 1R12: Maintenance Implementation

CRDRs

2682409, 2640655, 2547814, and 2691427

WOs

2584849, 2560952, 2551842, and 2584695

Miscellaneous

Low Pressure Alarm for Main Steam Isolation Valve MSIV-180 Train A Accumulator due to Faulty Pressure Transmitter (Unit 1, CRDR-2547756)

Steam Generator - Main Steam Isolation Valve Reliability Performance Criteria

Nuclear Administrative and Technical Manual 40OP-9SG01, "Main Steam," Revision 37

Nuclear Administrative and Technical Manual 90DP-0IP10, "Condition Reporting," Revision 18

Nuclear Administrative and Technical Manual 70DP-0MR01, "Maintenance Rule," Revision 9

Nuclear Administrative and Technical Manual 70DP-0EE01, "Equipment Root Cause of Failure Analysis," Revision 12

Section 1R13: Risk Assessment/Emergent Work

CRDRs

2704090, 2713743, and 2711884

Procedures

31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revision 15
31MT-9RC43, "Control Element Assembly Extension Shaft Replacement," Revision 8

Section 1R14: Nonroutine Events

CRDRs

2327899, 2707423, 2714544, 2717298, 2715941, and 2715727

Procedures

40EP-9EO02, "Reactor Trip," Revision 7
40OP-9ZZ10, "Mode 3 to Mode 5 Operation," Revision 42
42ST-2ZZ02, "Inoperable Power Sources Action Statement," Appendix B, Revision 31
78OP-9ZZ02, "NAC-UMS® Cask Loading Operations," Revision 8

Vendor Drawings

PVNGS-026, "VDS P&ID Schematics," Revision J
PVNGS-030, "Cooldown Elevations," Revision B
PVNGS-031, "VDS Spargers," Revision D

Miscellaneous

Core Protection Calculator Functional Block Diagram

Section 1R15: Operability Evaluations

CRDRs

2703945, 2705929, 2693663, 2704218, 2702302, 2707309, and 2707812

Procedures

Procedure 90DP-0IP1, "Condition Reporting," Revision 16
Procedure 40DP-9OP26, "Operability Determination," Revision 12

WOs

2693355, 2394073, and 2342097

Miscellaneous

Operability Determination 202, Revision 4

TSCCR 2375135, "LCO 3.6.3 Containment Isolation Valves SR 3.6.3.6 not Performed

Adequately for CP 2B and 3B"

Design Change Request 2455773, "Design and Install a 42" Testable Blind Flange Inside Containment on Penetrations 56 and 57 of the Containment Purge Refueling Train"

Analysis RA-13-C00-1997-052, "Proprietary-COLSS Cycle Independent Database Constants Analysis," Revision 15

Deficiency WO 2704890

Section 1R19: Postmaintenance Testing

CRDRs

2699288, 2699943, and 2700187

Engineering Source Document, DMWO 2692447, "
Appendix A to 01-J-ZZI-004, "Controlled Motor Operator Database (CMODB)," Revision 22

Procedures

30DP-9MP01, "Conduct of Maintenance," Revision 36

Drawings

01-E-SIF-011, "Control Wiring Diagram - Safety Injection Shutdown CLG System HPSI 2 Flow Control to Reactor Coolant Valve 1J-SIB-UV-626," Revision 2

01-E-SIB-011, "Elementary Diagram - Safety Injection Shutdown CLG System HPSI 2 Flow Control to React Coolant Valve 1JSIB-UV-626," Sheet 2, Revision 7

Section 1R20: Refueling and Outage Activities

CRDRs

2695262, 2710401, and 2707372

Procedures

40OP-9ZZ16, "RCS Drain Operations," Revision 40
40OP-9ZZ20, "Reduced Inventory Operations," Revision 5
40OP-9SI01, "Shutdown Cooling Initiation," Revision 32
40ST-9ZZ09, "Containment Cleanliness Inspection," Revision 8
72IC-9RX03, "Core Reloading," Revision 24

Drawings

01-M-SIP-001, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 25

01-M-SIP-002, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 24

01-M-RCP-001, "P&I Diagram, Reactor Coolant System," Revision 30

01-P-RCF-172, "Containment Building Reactor Vessel Head Vent Isometric," Revision 1

Vendor Drawing C-246-751-0, "Vent Pipe Orifice Installation," Revision 1

Vendor Drawing C-STD11-1070036-00, "Vent Pipe Orifice - 182.25" ID PWR," dated November 24, 1981

<u>Miscellaneous</u>

U1R11 Shutdown Risk Assessment, Revision 0
U1R11 Shutdown Risk Assessment, Revision 1
U1R11 Shutdown Risk Assessment, Revision 2

Component Data Sheet for Valve 1PRCEVV221
Clearance 102479, "EDG Storage Tank"
Clearance 99955, "Half Pipe Permit"

<u>Section 2OS1: Access to Radiologically Significant Areas</u>

<u>CRDRs</u>

2489355, 2616040, 2637565, 2639813, 2642822, 2646026, 2650039, 2651063, 2651332, 2655725, 2664593, 2665023, 2670009, 2689815, 2691798, 2695319, 2695508, and 2696454

<u>Nuclear Assurance Evaluation Reports and Self Assessments</u>

ER-03-0151, "Dosimetry, Electronic Logs, SGRP, Training"

ER-03-0266, "ALARA, Contamination Control, Surveys, Radiation Exposure Permits"

ER-03-0269, "Reactor Head Stand, Posting, Job Coverage"

ER-03-0281, "U2R11Posting, REP's, and Surveys"

ER-03-0289, "Posting, Labeling, RP Walkdowns, Pump Bay Worker Knowledge"

ER-03-0300, "Radworker Practices, ALARA, Job Coverage, Contamination Control, Radiography"

ER-03-0339, "Radioactive Material, Posting, Job Coverage, Radiography"

ER-03-0451, "Radiation Exposure, Access Control, Locked High Radiation Area
Self Assessment, Review of High Noise EPD Utilization in U2R11," dated January 22, 2004

Radiation Exposure Permits

1-1343A, "Refuel Machine- Hoist Box Maintenance"
1-3002E, "Reactor Destack and Restack"
1-3306E, "Primary Side Steam Generator Maintenance"
1-3047A, "Reactor Vessel Closure Head Insulation Modification and Inspection"

Procedures

75DP-9RP01, "Radiation Exposure and Access Control," Revision 6

75DP-0RP02, "Radioactive Contamination Control," Revision 5

75RP-0RP01, "Radiological Posting and Labeling," Revision 19

75RP-9OP01, "Radiological Controls for Diving Operations," Revision 7

75RP-9OP02, "Control of Locked High radiation Areas and Very High Radiation Areas," Revision 15

75RP-9RP02, "Radiation Exposure Permits," Revision 16

75RP-9RP10, "Conduct of R.P. Operations," Revision 12

75RP-9RP16, "Special Dosimetry," Revision 10

Section 1EO6: Drill Evaluation

04-D-FAC-06005, "2004 Emergency Preparedness Drill"

EPIP-01, "Satellite Technical Support Center Actions," Revision 15

EPIP-02, "Operations Support Center Actions," Revision 27

EPIP-03, "Technical Support Center Actions", Revision 33

EPIP-04, "Emergency Operations Facility Actions," Revision 33

EPIP-99, "Standard Appendices, Appendix A- Emergency Action Levels," Revision 1

EPIP-99, "Standard Appendices, Appendix B- Protective Action Recommendations," Revision 1

EPIP-99, "Standard Appendices, Appendix D- Notification," Revision 1

EPIP-99, "Standard Appendices, Appendix E- ERDS Activation," Revision 1

EPIP-99, "Standard Appendices, Appendix F- Dose Projection," Revision 1

EPIP-99, "Standard Appendices, Appendix G- Core Damage Assessment," Revision 1

EPIP-99, "Standard Appendices, Appendix H- Autodialer Activation," Revision 1

EPIP-99, "Standard Appendices, Appendix P- EAL Technical Bases," Revision 1

Section 4OA2: Identification and Resolution of Problems

CRDRs

2641696, 2456073, 2638327, 2630020, 2635803, 2639367, 2639681, 2643914, 2643868, 2643998, 2645590, 2647458, 2652045, 2653658, 2659887, 2684877, 2686659, 2687062, 2691715, 2699434, 2699031, 2699765, 2700949, 2705184, and 2706202

Procedure

01DP-9EM01, "Overtime Limitations," Revision 3

Section 4OA5: Other Activities

CRDRs

2699775, 2687861, and 2626902

Procedures

73TI-9ZZ78, "Visual Examination for Leakage," Revision 4
31ST-9SI01, "Cleaning/Inspection of ECCS Sumps," Revision 7
40ST-9ZZ09, "Containment Cleanliness Inspection," Revision 8

WO

02580581

Miscellaneous

10 CFR 50.59 Screening, "40ST-9ZZ09 Procedure Change for Inspection to Ensure Open Position of RCP Bay Personnel Access Gates," Revision 0

Calculation 01-EC-MA-0221, "AC Distribution," Revision 8

Calculation 02-EC-MA-0221, "AC Distribution," Revision 8

Calculation 03-EC-MA-0221, "AC Distribution," Revision 8

Calculation 13-EC-PB-0202, "Degraded Voltage Relay Setpoint," Revision 2

GL-79-36, "Adequacy of Station Electric Distribution Systems Voltages," http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1979/gl79036.html

- Millstone 7/5/1976
- ANO 9/16/78

NUREG-0800, Standard Review Plan, Branch Technical Position PSB-1, "Adequacy of Station Electric Distribution System Voltages," http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/ch8/08a-app.pdf

Information Notice IN 2000-06, Offsite Power Voltage Inadequacies, ADAMS Accession ML003695551

Regulatory Information Summary RIS 2000-24, Concerns about Offsite Power Voltage Inadequacies and Grid Reliability Challenges Due to Industry Deregulation, ADAMS Accession ML003695551

Callaway - LER 50-483/99-005, "Loss of Both Offsite Sources," ADAMS Accession ML003706314, ML 003684343, and ML003691949

LIST OF ACRONYMS

ADV	atmospheric dump valve
ASME	American Society of Mechanical Engineers
CEA	control element assembly
CFR	*Code of Federal Regulations*
CRDR	condition report/disposition request
ECCS	emergency core cooling system
LER	licensee event report
LOOP	loss of offsite power
MFW	main feedwater
RCS	reactor coolant system
RP	radiation protection
RPV	reactor pressure vessel
RTD	resistance temperature detector
SDC	shutdown cooling
SFP	spent fuel pool
TI	temporary instruction
URI	unresolved item
WO	work order