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

May 13, 2004

Greg R. Overbeck, Senior Vice
President, Nuclear
Arizona Public Service Company
P. O. Box 52034
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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000528/2004002, 05000529/2004002, AND 05000530/2004002

Dear Mr. Overbeck:

On March 31, 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 April 1 and 28, 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 one finding concerning high vibrations on high pressure safety injection piping. This finding has potential safety significance greater than very low significance. The finding did present an immediate safety concern. However, corrective actions were implemented to repair the piping and reduce the vibrations. In addition, the report documents one self-revealing and three inspector identified findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements; however, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations consistent with Section VI.A of the NRC Enforcement Policy. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-001; and the NRC Resident Inspector at Palo Verde Nuclear Generating Station, Units 1, 2, and 3, facility.

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

Sincerely,

/RA/

Troy W. Pruett, Chief
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Division of Reactor Projects

Dockets: 50-528
50-529
50-530
Licenses: NPF-41
NPF-51
NPF-74

Enclosure:
NRC Inspection Report 05000528/2004002, 05000529/2004002, and 05000530/2004002
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-528, 50-529, 50-530
Licenses:	NPF-41, NPF-51, NPF-74
Report No:	05000528/2004002, 05000529/2004002, and 05000530/2004002
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Tonopah, Arizona
Dates:	January 1 through March 31, 2004
Inspectors:	J. Anderson, Emergency Preparedness Specialist, Office of NRR E. Crowe, Resident Inspector, Arkansas Nuclear One D. Dumbacher, Project Engineer P. Elkmann, Emergency Preparedness Inspector, Operations Branch J. Hanna, Resident Inspector, Callaway T. Jackson, Resident Inspector, Diablo Canyon J. Kramer, Senior Resident Inspector, Fort Calhoun J. Melfi, Resident Inspector R. Mulliken, Senior Reactor Inspector N. Salgado, Senior Resident Inspector G. Warnick, Resident Inspector
Approved By:	Troy W. Pruett, Chief, Project Branch D Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000529/2004002, 05000528/2004002; 05000530/2004002; 1/01/04 - 3/31/04; Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Integrated Resident and Regional Report; Equip. Align., Main. Risk, Non-routine Evol., Other Activities, EAL and E-Plan Changes.

This report covered a 3-month period of inspection by resident inspectors, the Fort Calhoun senior resident inspector, resident inspectors from Diablo Canyon, Callaway, and Arkansas Nuclear One, a senior reactor inspector, a project engineer and two emergency preparedness specialists. The inspection identified four Green noncited violations and one apparent violation with potential safety significance greater than Green. 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-649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a because an inadequate work order was used to perform a pressurizer level control system data collection engineering action plan. The work order was inadequate in that it resulted in exceeding the maximum pressurizer level allowed by Technical Specification 3.4.9.

  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 reactor coolant system barrier from radionuclide releases caused by accidents or events. Using the Significance Determination Process Phase 1 Worksheet, 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. This issue involved human performance cross-cutting aspects associated with poor decision making (Section 1R13).

- TBD. A self-revealing apparent violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, was identified when an incorrect design configuration, combined with high vibrations, caused high cycle fatigue in a socket weld upstream of high pressure safety injection header drain Valve 1-P-SIA-V056, resulting in a reactor coolant system pressure boundary leak.

  The finding is greater than minor since it had an actual impact to the reactor coolant system boundary. Using the Significance Determination Process

Phase 1 and Phase 2 Worksheets, the finding was determined to affect both the barrier integrity and initiating events cornerstones. The finding was determined to have potential safety significance of greater than very low significance because of the possible failure mode of the piping and the duration of the degraded condition (Section 1R14).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation for the failure to comply with 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions, Specifically, the licensee did not identify the degradation of polyethylene insulating channels on Class 1E station batteries. Missing insulating channels could affect the seismic qualification of the batteries.

  This finding is greater than minor because it affects the reactor safety mitigating system cornerstone objective to ensure the capability of systems that respond to initiating events. Using the Significance Determination Process Phase 1 Worksheet, the finding was determined to have a very low safety significance, since there was no case where enough insulating channels had slipped to affect the seismic analyses, and the batteries remained in their design configuration. This issue involved problem identification and resolution cross-cutting aspects associated with the identification and degraded conditions (Section 1R04.2).

- Green. The inspectors identified a noncited violation for the failure to comply with Technical Specification 3.0.4 in that Mode 3 was entered on two occasions, once on December 8, 2003, and again on December 10 when compliance with Technical Specification 3.7.5, "Auxiliary Feedwater System," had not been established. Specifically, the acceptance criteria of Procedure 73ST-9XI38, "AFA-P01 Discharge Check Valve AFA-V015 - Inservice Test," was not met. Consequently, the required number of auxiliary feedwater trains were not available to support plant conditions in Mode 3.

  The finding is greater than minor since it is associated with the equipment performance attribute of the mitigating systems cornerstone and affects the cornerstone objective of equipment availability. Using the Significance Determination Process Phase 1 and 2 Worksheets, the finding was determined to effect the loss of a single train of a system for greater than its Technical Specification allowed outage time. The finding was determined to have very low safety significance because the exposure time for this condition was less than 2 days and all mitigation capabilities described on the selected Significance Determination Process Phase 2 worksheets for the applicable core damage sequences were maintained. This issue involved human performance cross-cutting aspects associated with poor decision making (Section 4OA5.1).

Cornerstone: Emergency Preparedness

- Green. On February 16, 2003, the licensee implemented an emergency plan change which decreased the required number of onshift emergency responders. This change constituted a decrease in effectiveness of the emergency plan because it could have resulted in a dedicated onshift communicator being replaced by a shift technical advisor, with a loss of one onshift position. Implementation of changes to the emergency plan, which constitute a reduction in the effectiveness of the plan without prior NRC approval, was a noncited violation of 10 CFR 50.54(q).

  The finding was evaluated using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," Section IV, because licensee reductions in the effectiveness of its emergency plan impact the regulatory process. The finding has greater than minor significance because reducing the required number of onshift emergency responders had the potential to impact the ability to perform all necessary emergency functions. The finding is determined to be a noncited Severity Level IV violation because the emergency plan change constituted a failure to implement a regulatory requirement, but did not constitute a failure to meet an emergency planning standard as defined by 10 CFR 50.47(b) because actual staffing levels remained above the emergency plan minimum (Section 1EP4).

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full power until February 3, 2004, when the reactor was shut down to repair reactor coolant system (RCS) leakage identified upstream of high pressure safety injection (HPSI) header drain Valve SIA-V056. The unit returned to essentially full power on February 9 following repairs to the RCS and remained there for the duration of the inspection period.

Unit 2 operated at full power until February 19, 2004, when the reactor was shut down due to the detection of a small primary to secondary tube leak on Steam Generator 1. On March 6, 2004, following repairs to the steam generator, unit start up was halted when a RCS leak was identified on a control rod drive mechanism vent valve assembly. The unit returned to Mode 5 and the vent valve assembly was repaired. The unit returned to full power on March 10, 2004, and remained there for the duration of the inspection period.

Unit 3 operated at essentially full power until February 28, 2004, when the main turbine generator tripped, immediately followed by an automatic reactor power cutback. The cause of the turbine trip was the loss of the generator field due to an excitation control failure. The unit was shut down to Mode 3 to troubleshoot and repair the excitation control problem. The unit commenced a cool down to Mode 5 on February 29, 2004, when a RCS leak was identified on one of the pressurizer heater sleeves. The unit returned to full power on March 9, 2004, following repairs to the main turbine generator and pressurizer heater sleeve and remained there for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), the Design Basis Manual, and other plant documents to verify that procedures and equipment are in place and maintained to prepare the licensee for anticipated hot weather conditions. The inspectors also performed a walkdown of the auxiliary building chillers and the control rod drive motor generator areas to identify any conditions which may adversely affect adverse weather protection preparedness.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

1. Partial Walkdown

a. Inspection Scope

The inspectors completed a partial walkdown of the four 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: 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.

- January 12, 2004, auxiliary feedwater (AFW) system Train B (Unit 1)

- February 18, 2004, HPSI (Unit 3)

- February 22, 2004, verification of reactor vessel level instrumentation alignment prior to midloop, per Procedure 40OP-ZZ16, "RCS Drain Operations," Revision 2, Appendix D (Unit 2)

- February 25, 2004, shutdown cooling Train B during midloop operations (Unit 2)

b. Findings

No findings of significance were identified.

2. Complete Walkdown

a. Inspection Scope

On January 26 and 27, 2004, the inspectors performed a complete system walkdown of accessible portions of the Vital 125 VDC system. During this walkdown, the inspectors verified electric power availability, labeling, hangers and support installation, and the status of associated support systems. Positions of electrical power breakers were compared to electrical drawings and piping and instrumentation drawings of the various support systems for the batteries. The inspectors also reviewed the status of outstanding WOs on the system.

b. Findings

Introduction. A finding was identified by the inspectors concerning cracked polyethylene insulating channels on the battery racks that affected the seismic qualification of the Class 1E station batteries.

Description. The inspectors noted numerous cracks in the polyethylene insulating channels that cover the battery rack steel next to the battery jars. The polyethylene insulating channels do not provide any strength to the rack structural steel, but prevent acid from degrading the steel and provide electrical isolation. This condition affected all four Class 1E batteries in all three units. The polyethylene channels were generally in place between the steel channels and the battery jars although some of the insulating channels had slipped slightly. The dimensions of the insulating channels were included in the seismic qualification of the batteries and racks. The operability determination performed for this condition noted that, if the polyethylene insulating channels were missing on opposite ends of a battery jar or rows of batteries, it would place them outside the assumptions in the seismic qualification report.

The racks were designed to have minimal spacing between the rails and the battery jar. The thickness of the insulating channel (approximately 1/8 inch) was taken into consideration during the design of the racks. The battery jars are in contact with the insulating channels. Double sided tape attaches the inner part of the channel to the structural steel to keep the channel in place. The vendor's gap tolerance is 0.25 inches. If channels on both sides of the battery were missing, the total gap between the rails and the battery jar could exceed the tolerance.

The UFSAR and the design basis note that the batteries are qualified as Seismic Class I. The Seismic Qualification Report C017090-1, "Battery Rack Seismic Qualification," specified that, the spacing of the steel in the racks required that the dimensions of the channels and batteries be maintained. However, the inspectors determined that the channels were not secured in a manner that could have prevented dislodging two or more channels around a battery jar.

Analysis. The finding adversely impacted the Class 1E battery qualification, because several channels could have slipped off of the battery rack. This finding is greater than minor because it affects the reactor safety mitigating system cornerstone objective to ensure the capability of systems that respond to initiating events. Using the Significance Determination Process Phase 1 Worksheet, the finding was determined to have a very low safety significance, since there was no case where enough insulating channels had slipped to affect the seismic analyses, and the batteries remained in their design configuration. The inspectors noted problem identification and resolution weaknesses in that operations and engineering personnel failed to recognize the degraded material condition of the insulating channels on the Class 1E station batteries during routine tours and system walkdowns.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Actions, states, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, licensee personnel did not identify the degradation of the insulating channels used on the Class 1E station batteries. Because the finding is of very low safety significance and has been entered into the corrective action program as CRDR 2667948, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528, 529, 530/2004002-01, "Failure to Identify Degradation of Polyethylene Channels on Class1E Batteries."

1R05 Fire Protection (71111.05)

.1 Routine Inspection

a. Inspection Scope

The inspectors conducted tours of the seven 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 fire damage from propagation of potential fires:

- January 12, 2004, diesel generator building, 131-foot, 115-foot, and 100-foot elevations (Unit 3)
- January 13, 2004, diesel generator building, 131-foot, 115-foot, and 100-foot elevations (Unit 2)
- January 13, 2004, diesel generator building, 131-foot, 115-foot, and 100-foot elevations (Unit 1)
- January 14, 2004, control building, 160-foot, 120-foot, 100-foot, and 74-foot elevations (Unit 1)
- January 30, 2004, auxiliary building, 140-foot, 120-foot, and 100-foot elevations (Unit 3)
- March 3, 2004, auxiliary building, 140-foot, 120-foot, and 100-foot elevations (Unit 2)
- March 15, 2004, condensate pump building and transfer tunnel (Unit 2)

b. Findings

No findings of significance were identified.

2. Fire Drill - Turbine Building 140-Foot Elevation (Unit 2)

a. Inspection Scope

On January 22, 2004, the inspectors observed the first quarter fire drill to evaluate the readiness of the licensee's personnel to prevent and fight fires. The inspectors reviewed the strategies and information in the Pre-Fire Strategies Manual, Revision 14, to verify that it accurately described the fire protection design features, fire area boundaries, and combustible loading for the lower cable spreading room. The inspectors observed the fire team enter the fire area and utilize the pre-fire plan strategies; the equipment brought to the scene to evaluate whether sufficient equipment was available for the simulated fire; and firefighting directions and radio communications between the fire commander, fire department personnel, and the control room. Also, the inspectors assessed the post drill critique to evaluate whether the drill acceptance criteria were satisfied.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the UFSAR, the Design Basis Manual, and other licensee documents to verify that the internal flood mitigation plans and equipment were consistent with the plants' design requirements and risk analysis assumptions. The inspectors also conducted walkdowns of two areas (main steam support structure and AFW rooms) on March 4-5, 2004. The inspectors verified, through direct observation and review of preventive maintenance records, that seals and dams between the rooms were maintained and the level switches in the AFW rooms were tested routinely.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On January 27, 2003, the inspectors observed operations crew performance during evaluated simulator Scenario SES0-08-C-00, "Pressurizer Instrument Failure, Loss of Running TC Pump, ATWS, LOOP, Blackout." 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 Implementation (71111.12)

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":

- Battery postcorrosion detected on several PK batteries as documented in CRDR 2399780 (Unit 3)
- Failure of close springs for emergency diesel generator output Breaker 3MDGAH01 to recharge as documented in CRDR 2660246 (Unit 3)

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 nine 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. The specific activities reviewed were associated with planned and emergent maintenance on:

- January 13, 2004, troubleshooting activities for control element drive mechanism Fans A and C per Work Mechanism (WM) 2664667 (Unit 2)
- January 23, 2004, replacement of a failed electro hydraulic control power supply per WM 2665898 (Unit 3)
- January 26, 2003, troubleshooting activities for replacement of hydrogen recombiner Train A trickle heater indicating bulb during performance of Procedure 36ST-9HP01, "Hydrogen Recombiner Instrumentation Calibration and Functional Test," Revision 8 (Unit 1)

- January 29, 2004, replacement of a loading rate board on the electro hydraulic control load control circuit, per WO 2662025 (Unit 2)

- January 29, 2004, evaluated licensee's assessment and troubleshooting of increasing containment sump inleakage trend (Unit 2)

- February 3, 2004, pressurizer level control system data collection per WO 2664844 (Unit 2)

- February 4, 2004, RCS freeze seal to repair weld on HPSI header drain Valve 1-P-SIA-V056 per Procedure 33MT-9ZZ02, "Freeze Sealing," Revision 5 (Unit 1)

- February 24-26, 2004, inability to properly install steam generator nozzle dams per Procedure 31MT-9RC48, "Steam Generator "NES" Nozzle Dam Installation and Removal," Revision 23 (Unit 2)

- March 9, 2004, performance of Procedure 40ST-9EC03, "Essential Chilled Water & Ventilation Systems Inoperable Action Surveillance," Revision 12, Section 8.19, for inoperable essential safeguard features switchgear room air handling unit (Unit 3)

b. Findings

Introduction. A finding was identified when an inadequate WO was used to perform a pressurizer level control system data collection engineering action plan. The WO was inadequate in that its implementation resulted in exceeding the maximum pressurizer level allowed by Technical Specification 3.4.9.

Description. On February 3, 2004, the licensee was implementing WO 2664844 to collect inservice data to address and optimize letdown oscillations per an engineering action plan. The intent of the action plan was to present a method for determining optimal controller settings to minimize letdown oscillations while minimally impacting plant operations.

10 CFR 50.59 Screening S-04-0006, Revision 0, was performed to review the proposed activity. The underlying assumption used in the screening to justify WO implementation was that pressurizer level would be maintained within the limits required by Technical Specifications. Restrictions and guidelines were discussed in the basis for the screening justification to provide additional margin to ensure adherence to Technical Specification level requirements. These included lowering pressurizer level "low in the band" prior to initiating each level increase and maintaining level within the programmed band. The WO to perform the work included a precaution to maintain level within the limits of Technical Specification LCO 3.4.9, but did not mention any requirements to maintain level within the programmed band. Additionally, the requirement to have pressurizer level low in the band (approximately 35 percent) was included as an initial condition to the entire data collection action plan rather than specifying level to be low in

the band prior to each level change. The licensee did not have a process to ensure that underlying assumptions identified in the safety screening were properly incorporated into the WO. Consequently, pressurizer level continued to increase throughout the data collection evolution and was allowed to exceed the programmed band upper limit of 52.6 percent on three occasions. During one iteration, operators induced a 4 percent increase on the controller from an initial pressurizer level of 47 percent. The anticipated 3 percent overshoot was greater than expected and level finally turned at approximately 57 percent. Pressurizer level exceeded the Technical Specification LCO limit of 56 percent for 10 minutes.

Analysis. The deficiency associated with this event is an inadequate procedure which led to exceeding the maximum allowed pressurizer level requiring entry into Technical Specification 3.4.9, Condition A. 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 reactor coolant system barrier from radionuclide releases caused by accidents or events. Using the Significance Determination Process Phase 1 Worksheets, 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. The inspectors noted human performance weaknesses associated with operations personnel decision making prior to, and following, the upper pressurizer level limit overshoot. Specifically, operators failed to recognize that implementation of the inadequate procedure established plant conditions where exceeding the Technical Specification limit was likely.

Enforcement. Technical Specification 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 9a, requires maintenance that can affect safety-related equipment to be properly preplanned and performed in accordance with written instructions appropriate to the circumstances. Contrary to the above, the licensee failed to include critical information in the WO to establish initial conditions, precautions, and limitations assumed in the 10 CFR 50.59 screening to ensure that pressurizer level was maintained within Technical Specification limits. Because the finding is of very low safety significance and has been entered into the corrective action program as CRDR 2669486, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2004002-02, "Pressurizer Level Transient Above Technical Specification Limit."

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

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

- On February 3, 2004, while performing the initial walkdowns to install a temporary modification in containment, plant personnel observed leakage from the insulation around HPSI header drain Valve SIA-V056, a 1-inch drain valve near shutdown cooling suction isolation Valve SIA-V651. Plant personnel discovered an unisolable RCS pressure boundary pinhole leak on the upstream weld of the drain valve. The licensee shutdown Unit 1 as required by Technical Specification 3.4.14.

  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 2669474.

- On February 19, 2004, the Unit 2 control room received an unexpected alarm on their radiation monitors indicating a tube leak on Steam Generator 1. Operators entered Abnormal Operating Procedure 40A0-9ZZ02, "Excessive RCS Leakrate," and identified a 3-5 gallon per day primary to secondary leak. An orderly power reduction commenced and the plant was shutdown to isolate Steam Generator 1.

  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 2 event was documented in CRDR 2685303.

- On February 28, 2004, Unit 3 experienced a main generator trip and a reactor power cutback. The operators had been experiencing excitation system trouble alarms the previous night. At 7:38 a.m., the main generator tripped from approximately 99 percent power due to a loss of field. The reactor received a reactor power cutback as designed. Operators stabilized the reactor at approximately 40 percent power. Based on the main generator problem, operations management decided to shutdown the reactor.

  The inspectors responded to the control room to evaluate 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 2687145.

b. Findings

Introduction. A Green self-revealing apparent violation was identified when an incorrect design configuration, combined with high vibrations, caused high cycle fatigue in a socket weld upstream of HPSI header drain Valve 1-P-SIA-V056, resulting in a small RCS pressure boundary leak.

Description. On February 3, 2004, during preparations for a temporary modification in containment, plant personnel noticed a leak coming from a socket weld on the upstream (unisolable) side of drain Valve 1-P-SIA-V056. The drain line connected to the 3-inch HPSI header that attaches to the shutdown cooling suction line.

The leak was due to a high cycle fatigue crack in the weld. The licensee noted that a piping support hanger configuration for this valve did not match the design configuration per Design Modifications 1SM-XM-001 and 1SM-XM-002. The design modifications provided supports for root valves on Class I lines in August 1990. A fixed support hanger on the HPSI header drain line was supposed to have been removed from Valve 1-P-SIA-V056 following the installation of a support to connect the drain line to the HPSI header. Since the original hanger that supported the drain line remained in place, stresses were concentrated near the original support. This stress concentration led to cyclic fatigue on the weld resulting in the RCS pressure boundary leak.

Analysis. The finding has potential safety significance greater than Green. The finding is greater than minor since it had an actual impact to the reactor coolant system boundary. Using the Significance Determination Process Phase 1 and Phase 2 Worksheets, the finding was determined to affect both the barrier integrity and initiating events cornerstones. The finding has potential safety significance greater than Green because the condition existed for greater than 3 days and a break of the piping could have exceeded 100 gpm. This issue was provided to a Senior Reactor Analyst for additional review.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, Design Control, states, in part that, measures shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee failed to implement Design Modifications 1SM-XM-001 and 1SM-XM-002 fully, in that a support that should have been removed was left in place. Pending determination of the finding's safety significance, this finding is identified as an apparent violation: AV 05000528/2004002-03, "Failure to Remove Pipe Support Leads to RCS Pressure Boundary Leak."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors evaluated the seven 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 were maintained or that Technical Specification actions were properly entered.

- December 8, 2003, operability determination associated with CRDR 2654231, "Essential Cooling Water Heat Exchanger Fouling due to Zinc Precipitate," Revision 1 (Units 1, 2, and 3)

- January 21, 2004, Operability Determination 270, "Can the Emergency DG be Considered Operable if the Local DG Annunciator Panel has the Alarms Locked in due to a Power Supply Failure" (Units 1, 2, and 3)

- January 26, 2004, Operability Determination 266, "Shutdown CEAs at Less Than UEL May Not Meet the Requirements Used in the Accident Analysis," Revision 1 (Units 1, 2, and 3)

- January 28, 2004, assessed compliance with Technical Specification 3.7.3, "Main Feedwater Isolation Valves," Condition A, when feedwater isolation Valve 137 did not move during performance of Procedure 73ST-9XI16, "Economizer FWIVs - Inservice Test," Revision 18 (Unit 2)

- January 31, 2004, assessed the impact of an erroneous computer code associated with core protection calculators that caused pretrip alarms during performance of Procedure 40ST-9SF01, "CEA Operability Checks," Revision 12 (Unit 2)

- February 19, 2004, Operability Determination 273, "Core Protection Calculator Software Design Error for CEA Position Indication" (Units 1 and 3)

- February 25, 2004, air entrainment in the shutdown cooling system during extended midloop operations documented in CRDR 2686273 (Unit 2)

b. Findings

Issues associated with air entrainment in the shutdown cooling system are documented in NRC Special Inspection Report 05000529/2004009. No other findings of significance were identified.

1R16 Operator Work-Arounds (711111.16)

a. Inspection Scope

The inspectors conducted interviews with operators, operator workaround program managers, and quality assurance personnel. The inspectors reviewed Units 1, 2, and 3 control room deficiency and operator challenges tracking lists to determine the number of operator workarounds that existed and to assess the cumulative effect of the workarounds, especially how these operator workarounds could adversely affect the operators' ability to respond to plant transients and/or accidents in a timely manner.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and evaluated the results from the following six 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.

- January 27, 2003, reviewed retest associated with Battery Charger BD voltage swing troubleshooting per WO 2627896 (Unit 2)
- January 2, 2004, replacement of valve actuator electric motor on low pressure safety injection Valve 1JSIBUV0625 (Unit 1)
- January 13-14, 2004, Battery Charger 3EPKAH15 inspection and replacement of circuit breaker starter coil (Unit 3)
- February 19, 2004, stroke time test of CPA-UV-4A/4B per Procedure 73ST-9XI15, "CP (Power Access Purge) Valves - Inservice Test," Revision 3 (Unit 2)
- February 29, 2004, repair of lower oil bearing on the low pressure safety injection Pump B per WM 2687307 (Unit 2)
- March 5, 2004, mechanical nozzle seal assembly installation on pressurizer heater Sleeve A03, per Deficiency WO 2687284 (Unit 3)

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

During the Unit 2 outage required to repair a steam generator tube leak, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

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

- 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, including switchyard activities, to ensure that Technical Specifications and outage safety plan requirements were met

- Monitoring of decay heat removal processes

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

- Startup and ascension to full power operation, tracking of startup prerequisites, and a walkdown of containment to verify that debris had not been left which could block emergency core cooling system suction strainers

- Licensee identification and resolution of problems related to outage activities

b. Findings

Issues associated with extended midloop operations are documented in NRC Special Inspection Report 05000529/2004009. No other findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

Applicable test data was reviewed to verify whether the licensee met Technical Specification, UFSAR, 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 eight surveillance tests:

- January 8, 2003, Procedure 73ST-9SI10, "HPSI Pumps Miniflow - Inservice Test," Section 8.2, Revision 27 (Unit 1)

- January 15, 2004, Procedure 32ST-9PK01, "7-Day Surveillance Test of Station Batteries," Revision 25 (Unit 3)

- January 15, 2004, Procedure 32ST-9PK02, "92-Day Surveillance Test of Station Batteries," Revision 27 (Unit 3)

- January 26, 2004, Procedure 72ST-9RX02, "Moderator Temperature Coefficient At Power," Revision 19 (Unit 2)

- February 18, 2004, Procedure 36ST-9SA05, "FBEVAS, CREVAS, & CRVIAS 18 Month Functional Test," Revision 8 (Unit 3)

- February 25, 2004, Procedure 73TI-9ZZ32, "Steam Generator Secondary Pressurization Test," Revision 6 (Unit 2)

- February 27, 2004, Procedure 33ST-9HJ02, "Surveillance Testing of the Control Room Nuclear Air Treatment System," Revision 7 (Unit 3)

- February 27, 2004, Procedure 73ST-9EC01, "Essential Chilled Water Pumps - Inservice Test," Revision 14, Section 8.2 (Unit 3)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors evaluated the following three temporary modifications and the associated 10 CFR 50.59 screening. 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. The inspectors observed the installation of and reviewed documentation for the following temporary modifications:

- January 22, 2003, T-Mod 2659140, "Temporary Setpoint Change for Reactor Vessel Head Seal Pressure Alarm Switch 2JRCNPSH118," Revision 0 (Unit 2)

- February 24, 2003, T-Mod 2654221 "Temporary Heater Blankets for the Heating of Piping Adjacent to shutdown cooling suction isolation Valve SI-UV-651," Revision 0 (Unit 1)

- March 2, 2004, Deficiency Work Order 2687350, "Removal of Damaged Copper Sheets From Main Generator Flexible Link and Installation of Insulating Tape to Secure Laminations," Revision 0 (Units 2 and 3)

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The inspectors evaluated the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The alert and notification system testing program was evaluated against the criteria contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, Federal Emergency Management Agency Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants," and the licensee's current Federal Emergency Management Agency approved alert and notification system design report. The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors reviewed emergency plan implementation procedures and other documents related to the emergency response organization augmentation system to determine the licensee's ability to staff emergency response facilities in accordance with their emergency plan and the requirements of 10 CFR Part 50, Appendix E. The inspectors also reviewed the documents pertaining to the installation and preoperational testing of an automated telephone dialing system.

The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors performed an inoffice review of Emergency Plan Implementation Procedure EPIP-99, "Standard Appendices," Revision 1, Appendix A, "Emergency Action Levels," submitted December 16, 2003. This revision lowered entry conditions based on high-range containment radiation monitors in Emergency Action Levels 1-4 and 1-11, and corrected references to the Offsite Dose Calculation Manual in Emergency Action Levels 3-3 and 3-10. The inspectors also reviewed the licensee's 50.54(q) evaluation of Emergency Plan, Revision 28, implemented February 2003. The revisions were compared to the previous revisions, to the criteria of NEI 99-01, "Methodology for Development of Emergency Action Levels," and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the revisions decreased the effectiveness of the emergency plan.

The inspectors completed one sample during this inspection.

b. Findings

Introduction. The inspectors determined that the licensee implemented a change to the numbers and duties of the dedicated onshift communicators and shift technical advisors, which constituted a decrease in effectiveness of the emergency plan. The licensee changed a requirement to have three dedicated communicators to allow one of three shift technical advisors to perform the communicator function as a collateral duty, reducing the required number of dedicated communicators to two. This finding was determined to be a noncited Severity Level IV violation of 10 CFR 50.54(q).

Description. The licensee implemented Revision 28 of their Emergency Plan on February 16, 2003. As part of this revision, the licensee revised note 1 to Table 1 (the onshift staffing table) and added note 1 to the shift technical advisor position. The revised note 1 permitted one shift technical advisor to perform the function of one required satellite technical support center communicator as a collateral duty. Emergency plan implementing procedures were changed to direct a shift technical advisor to fill the communicator function. This effectively permitted deletion of a dedicated person from the onshift emergency response roster.

Analysis. Implementation of an emergency plan change, which decreased the effectiveness of the emergency plan, was a performance deficiency. The finding was associated with a violation of NRC requirements. The finding had a credible impact on the emergency preparedness cornerstone objective because it represented a decrease in effectiveness of the licensee's emergency plan associated with emergency planning standard 10 CFR 50.47(b)(2). This finding is more than minor because a reduction in the number of onshift personnel could adversely impact the ability to perform necessary emergency functions. In accordance with Manual Chapter 0609, Appendix B, Sections 2.2(e) and 4.4, the inspectors evaluated the significance of the finding using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions (Enforcement Policy)," Section IV, "Significance of Violations." The finding was determined to be a Severity Level IV violation according to NUREG-1600 Supplement VIII, "Emergency Preparedness," because a reduction in the number of required onshift emergency response personnel affected the ability to implement an emergency planning standard, although the change was not a failure to meet Emergency Planning Standard, Section 50.47(b)(2), because the communicators were not actually removed from onshift crews.

Enforcement. 10 CFR 50.54(q) states, in part, ". . . . The nuclear power reactor licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E to this part. . . . ." Licensee implementation of changes to the emergency plan, which decreased the effectiveness of the emergency plan without prior NRC approval was a violation of Section 50.54(q). This violation is being treated as a noncited Severity Level IV violation consistent with Section VI.A of the Enforcement Policy. The finding is not suitable for significance determination process evaluation, but has been reviewed by NRC management and is determined to be a Green finding of very low safety significance. This issue has been entered into the licensee's corrective action program as CRDR 2670023. (NCV 05000528, 529, 530/2004002-04, Implementation of a Change to Table 1 which was a Decrease in Effectiveness of the Emergency Plan)

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed a summary of all CRDRs associated with emergency preparedness generated between July 1, 2002, and January 31, 2004, to determine the licensee's ability to identify and correct problems in accordance with the requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E. The inspectors also reviewed 1 root cause analysis, 4 drill reports, 7 self-assessments, 8 quality assurance surveillances and audits, and 24 specific CRDRs. Corrective actions were evaluated against the requirements of Procedure 90DP-0IP0, "Condition Reporting," Revision 16. Root cause analyses were evaluated against the requirements of the "Root Cause Investigation Manual for Significant CRDRs," Revision 2. The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed portions of the announced emergency preparedness drill conducted on February 11, 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.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Initiating Events Cornerstone

a. Inspection Scope

The inspectors reviewed unit logs, plant thermal performance records, control room logs, monthly operating reports, and licensee event reports (LERs) from January 2003 to December 2003 for all three units to verify the accuracy and completeness of data used to calculate and report the following performance indicators:

- Unplanned scrams per 7,000 critical hours
- Scrams with loss of normal heat removal
- Unplanned power changes per 7,000 critical hours

b. Findings

No findings of significance were identified. The performance indicators all remained in the licensee response band (Green).

Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors reviewed control room logs and LERs from January 2003 to December 2003 for all three units to verify the accuracy and completeness of data used to calculate and report the following performance indicator:

- HPSI Unavailability

b. Findings

No findings of significance were identified. The performance indicator remained in the licensee response band (Green).

Emergency Preparedness Cornerstone

a. Inspection Scope

The inspectors sampled submittals for the performance indicators listed below for January 1, 2003, through December 31, 2003. The definitions and guidance of

NEI 99-02, "Regulatory Assessment Indicator Guideline," were used to verify the licensee's basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period.

- Drill and exercise performance
- Emergency response organization participation
- Alert and notification system reliability

The inspectors reviewed a 100 percent sample of drill and exercise scenarios, licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspectors reviewed a 100 percent sample of emergency response organization participation data, as well as selected emergency responder qualification, training, and drill participation records. The inspectors reviewed alert and notification system maintenance records and procedures, and a 75 percent sample of siren test results. The inspectors also interviewed licensee personnel that were responsible for collecting and evaluating the performance indicator data. The inspectors completed three samples during this inspection.

b. Observations

The inspectors determined that four notification opportunities during operator simulator drills could not be verified as accurate because the licensee had not established clear thresholds for determining the status of a radiological release. Although the licensee's definition of a "release in progress" was radioactive material entering the environment as a result of events associated with an emergency for steam generator tube leak scenarios, the licensee had not clearly established a means for the control room to determine that a release condition existed or to characterize whether a release was above or below offsite dose calculation manual limits. The inspectors determined that licensee evaluators in the simulator used radiation monitor alarms as a de facto release threshold although this was not documented and was not consistent with the licensee definition of "release."

The inspectors determined that approximately 150 notification opportunities associated with individual performance drills for control room communicators could not be verified as accurate. Although each individual offsite notification form and evaluator record form was archived, the licensee did not present a standard scenario and the inspectors could not verify that each individual correctly recorded the information presented during the scenario.

NEI 99-02, "Regulatory Assessment Performance Indicator Guidelines," Revision 2, requires that the licensee designate those drills intended to be evaluated for the drill and exercise performance indicator in advance of the drills. The inspectors determined that the licensee failed to clearly designate 31 operator simulator drills and approximately 150 communicator drills as evaluated for the performance indicator prior to conducting the drills. The inspectors concluded that this failure to comply with NEI guidance did not affect evaluation of the drills but represented a deviation from the guidance and from licensee procedures.

The inspectors noted three examples of key responders counted in the emergency response organization drill participation performance indicator although licensee training records showed the individuals as not current in their emergency response positions. The inspectors noted one example of a responder not counted in the drill participation performance indicator when training records showed the individual should have been included. The inspectors verified that correction of the identified deviations would not cause the performance indicator to cross a significance threshold. The inspectors also identified seven examples of date errors in the tracking database; the most common example was recording of the incorrect year (e.g. 2004 or 2005 instead of 2003).

4OA2 <u>Problem Identification and Resolution</u>

1. <u>Routine Review of Identification and Resolution of Problems</u>

a. <u>Inspection Scope</u>

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 conditions significantly 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. <u>Findings</u>

No findings of significance were identified.

2. <u>Emergency Preparedness Annual Sample Review</u>

a. <u>Inspection Scope</u>

The inspectors reviewed performance and facility problems documented in Calendar Years 2002 and 2003 in the licensee’s corrective action program. The inspectors selected 20 CRDRs for detailed review based on their impact on the risk significant planning standards, emergency worker protection, and the ability to staff and maintain emergency response facilities. The selected corrective actions were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, appropriate corrective actions were identified and prioritized, and that effective corrective actions were completed. The inspectors evaluated the CRDRs against the requirements of Procedure 90DP-0IP0, "Condition Reporting," Revision 16.

b. Findings and Observations

No findings of significance were identified.

3. Cross-References to PI&R Findings Documented Elsewhere

Section 1R04.2 describes a finding where operations and engineering personnel failed to identify the degraded material condition of the insulating channels on the Class 1E station batteries during routine tours and system walkdowns.

4OA3 Event Followup

1. Shutdown Cooling System Air Entrainment (71153)

a. Inspection Scope

Evaluated plant conditions, equipment performance, and licensee actions related to shutdown cooling system air entrainment during midloop operations (Unit 2).

b. Findings

A special inspection was performed to review extended midloop operations and a steam generator tube leak event. The results will be documented in NRC Special Inspection Report 05000529/2004009.

2. (Closed) LER 05000530/2003001-00: Main Steam Safety Valve As-Found Lift Pressures Outside of Technical Specification Limits

On March 20, 2003, prior to the upcoming Unit 3 refueling outage, surveillance testing revealed that as-found lift pressure for one main steam safety Valve 3JSGEPSV0578 was greater than Technical Specification limits. The inspectors reviewed CRDR 2592898 and its significant root cause investigation. The licensee concluded the valve experienced bonding between the disc and nozzle which when coupled with temperature effects on the yoke rods and springs resulted in the valve lifting above its Technical Specification limit. The licensee's analysis found that the as-found condition of the Unit 3 main steam safety valve would not, under accident conditions, have resulted in peak pressures that would have exceeded the overpressure protection limits for the primary and secondary systems. No new findings were identified in the inspector's review. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

1. Cross-References to Human Performance Findings Documented Elsewhere

Section 1R13 describes a finding for an inadequate procedure and poor operations decision making that resulted in exceeding the upper pressurizer level Technical Specification limit.

Section 4OA5 of this report, and Section 1R22 of NRC Inspection Report 05000528/2003005, 05000529/2003005, and 05000530/2003005, described a finding that involved poor decision making by operations and engineering personnel for entering Mode 3 when operability of AFW pump Train B had not been confirmed.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000529/2003005-02: AFW Discharge Checkvalve Test Failure

Introduction. A Green noncited violation was identified for the failure to comply with Technical Specification 3.0.4. Specifically, Mode 3 was entered without two operable motor driven AFW trains as required by Technical Specification LCO 3.7.5.

Description. The inspectors identified that Mode 3 was entered on two occasions, once on December 8, 2003, and again on December 10 when compliance with Technical Specification 3.7.5 had not been established. Specifically, the acceptance criteria of Surveillance Procedure 73ST-9XI38, "AFA-P01 Discharge Checkvalve AFA-V015 Inservice Test," were not met. Consequently, the required number of AFW trains were not available to support plant conditions in Mode 3. On December 10, 2003, Unit 2 was in Mode 3 for 26.3 hours, a period greater than allowed by Technical Specification 3.0.4, before Procedure 73ST-9XI38 was successfully performed to establish operability of AFW Pump B.

NRC Inspection Report 05000528/2003005, 05000529/2003005, and 05000530/2003005 documents the justification licensee personnel used to enter Mode 3 with unacceptable surveillance test results. The inspectors completed their assessment of licensee performance for the Mode 3 entries through review of the evaluation documented in CRDR 2657316. The licensee concluded in the evaluation that the valve disk was dislodged from the sealed position prior to performance of the surveillance test on December 8, 2003. An assumed set of conditions were used in Calculation 02-MA-AF-0041 to establish that Valve AFA-V015 was closed but not tightly seated. Further, the evaluation concluded that the most likely reason for the lack of a tight seal on the valve was the result of operating AFW Pump A in the recirculation mode on November 20, 2003. The licensee assumed that this pump run caused the valve disc to slightly swing open, however, there was not enough momentum force to achieve a tight seal upon reseating of the disc. Based on these assumptions, the licensee further concluded that the valve would have fully seated with an increase in differential pressure and would have been capable of performing its safety function in Mode 3.

Following the evaluation review, the inspectors concluded that, although the licensee's assumptions and conclusions may have been correct, the operability justification for AFW Pump B could not eliminate the possibility that foreign material in the valve seat was the cause of the unacceptable surveillance test results. Foreign material in the valve seat could have prevented the valve from fully seating if required in Mode 3. Therefore, the surveillance test should not have been invalidated and operability of Valve AFA-V015 should have been established prior to entering Mode 3.

Analysis. The inspectors determined that the finding was greater than minor since it was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone objective of equipment availability. Using the Significance Determination Process Phase 1 Worksheet, the finding was determined to effect the loss of a single train of a system for greater than its Technical Specification allowed outage time. The Senior Reactor Analyst (SRA) determined that revision to the Phase 2 Significance Determination Process workbook was necessary for completion of the analysis. The condition existed for less than 2 days following a refueling outage. Therefore, the worksheets associated with transients and transients without the power conversion system were not used. The SRA determined that the decay heat rate for all events was very low due to the low operating modes of the facility following the refueling outage. Therefore, the time available to recover and align plant equipment could be extended. Lastly, the failure of the motor driven auxiliary feedwater pump would also require a simultaneous failure of the turbine driven auxiliary feedwater pump. The likelihood of two independent failures also reduced the safety significance of this finding. Because of the above factors, the SRA determined that the finding is of very low safety significance. The inspectors noted human performance weaknesses as described in this report, and Section 1R22 of NRC Inspection Report 05000528/2003005, 05000529/2003005, and 05000530/2003005, associated with licensee decision making that preceded the Technical Specification 3.0.4 violation.

Enforcement. Technical Specification 3.0.4, specified, in part, when a Limiting Condition for Operation is not met, entry into a Mode shall not be made except when the associated actions to be entered permit continued operation in the mode for an unlimited period of time. Contrary to the above, the licensee entered Mode 3 from Mode 4 on December 8 and 10, 2003, without the minimum number of operable motor driven AFW trains as required by Technical Specification 3.7.5. Because this violation is of very low safety significance and has been entered into the corrective action program as CRDR 2657316, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2004002-05, Failure to Establish AFW Pump Operability Prior to Mode 3 Entry.

2. (Closed) URI 05000528, 529, 530/2003007-01: Manual Actions Taken in Lieu of Physical Protection Requirements

a. Inspection Scope

The inspectors identified an URI concerning the use of manual actions for a fire outside of the control room. These manual actions were used in lieu of physical protection of equipment required for postfire safe shutdown and some of these actions did not appear to be approved by the NRC. This issue was made unresolved pending further NRC review of the licensee's fire protection program licensing basis regarding the use of manual actions.

In reviewing Procedure 40DP-9ZZ19, "Operational Considerations due to Plant Fire," and Calculation 13-MC-FP-316, "Appendix R, Manual Action Feasibility," the inspectors found that in the event of a fire in numerous fire areas outside of the control room, the licensee credited the use of manual actions in lieu of providing physical protection of equipment required for postfire safe shutdown. Many of these manual actions were found to have been submitted to the NRC and approved in the UFSAR, Safety Evaluation Report, Supplement 7, dated December 1984. However, the inspectors could not verify that all manual actions taken in lieu of providing the required physical protection were formally submitted to and approved by the NRC.

Subsequently, the inspectors performed an inoffice review of the licensing basis. The use of additional manual actions was provided to the NRC in the licensee's response to Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers," dated December 17, 1992. The licensee was using thermo-lag material as a fire barrier. However, in response to the generic letter the licensee proposed using manual actions to replace the physical protection that the thermo-lag material had previously provided. In a letter dated June 11, 1998, from the Office of Nuclear Reactor Regulation to the licensee, the NRC determined that the licensee response and actions in regard to Generic Letter 92-08 were complete. The inspectors determined that the use of manual actions in lieu of the physical protection requirements of 10 CFR Part 50, Appendix R, Section III.G.2 had been approved by the NRC.

The inspectors had previously reviewed the manual actions credited in the event of a fire in the selected fire areas and found that all were described in procedures and appeared to be reasonable and feasible.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The inspectors discussed the Emergency Protection preliminary inspection results with Mr. G. Overbeck, Senior Vice-President, Nuclear, and other members of his staff during an onsite debrief February 5, 2004, and final inspection results during a February 18, telephonic exit meeting.

The inspectors presented the inspection results to Mr. G. Overbeck, Senior Vice President, Nuclear, and other members of licensee management during an exit meeting conducted on April 1, 2004.

The inspections provided clarifying information associated with the resident inspector inspection results to Mr. G. Overbeck, Senior Vice-President, Nuclear, and other members of his staff during an exit meeting conducted on April 28, 2004.

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

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Allison, Program Advisor, Operations Training
T. Barsuk, Senior Coordinator, Emergency Preparedness
S. Bauer, Department Leader, Regulatory Affairs
D. Carnes, Director, Regulatory Affairs and Nuclear Assurance
D. Crozier, Group Leader, Emergency Preparedness
E. Dutton, Section Leader, Performance Improvement
J. Hesser, Director, Emergency Services
D. Kanitz, Regulatory Affairs
D. Marks, Section Leader, Regulatory Affairs - Compliance
D. Mauldin, Vice President, Engineering and Support
G. Overbeck, Senior Vice-President
T. Radtke, Director, Operations
M. Renfroe, Section Leader, Design Engineering
F. Riedel, Director, Nuclear Training Department
J. Scott, Department Leader, Nuclear Assurance
D. Smith, Plant Manager, Production
M. Sontag, Department Leader, Nuclear Assurance
M. Van Dop, Department Leader, System Engineering
P. Wiley, Department Leader, Operations Training
J. Wood, Section Leader, Operations Training

Others

F. Gowers, Site Representative, El Paso Electric
R. Henry, Site Representative, Salt River Project

NRC Personnel

M. Satorius, Deputy Directory, Division of Reactor Projects (DRP)
J. Clarke, Senior Project Engineer, DRP

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000528, 529, 530/2004002-01	NCV	Failure to Identify Degradation of Polyethylene Channels on Class 1E Batteries (Section 1R04.2)
05000529/2004002-02	NCV	Pressurizer Level Transient Above Technical Specification Limit (Section 1R13)

05000528, 529, 530/2004002-04	NCV	Implementation of a Change to Table 1 which was a Decrease in Effectiveness of the Emergency Plan (Section 1EP4)
05000529/2004002-05	NCV	Failure to Establish AFW Pump Operability Prior to Mode 3 Entry (Section 4OA5.1)

Opened

05000528/2004002-03	AV	Failure to Remove Pipe Support Leads to RCS Pressure Boundary Leak (Section 1R14)

Closed

05000529/2003005-02	URI	AFW Discharge Checkvalve Test Failure (Section 4OA5.1)
05000528, 529, 530/2003007-01	URI	Manual Actions Taken in Lieu of Physical Protection Requirements (Section 4OA5.2)
05000530/2003001-00	LER	Main Steam Safety Valve As-Found Lift Pressures Outside of Technical Specification Limits (Section 4OA3.2)

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

WO
2664515

Drawings

Vendor Drawing 017090-BM-1, "Bill of Materials for 32-Cell Battery Rack," Revision 5

Vendor Drawing 017090-CD-1, "Connection Diagram, Room A, Units 1, 2 & 3," Sheet 1 of 4, Revision 2

Vendor Drawing 017090-GA-1, "General Arrangement 32-Cell Battery Rack," Sheet 1 of 3, Revision 3

Vendor Drawing 017090-GA-1, "General Arrangement 32-Cell Battery Rack," Sheet 3 of 3, Revision 4

Vendor Drawing 017090-ME-1, "Battery Racks - Item No. 1 Mechanical Details," Revision 4

13-E-ZJP-001, "Battery and DC Equip Rooms Plan," Sheet 1 of 2, Revision 12

01-E-PKA-001, "Main Single Line Diagram, Class 1E 125V DC Class 1E and 120 V AC Inst Power System," Revision 4

01-M-HJP-002, "Control Building HVAC P&I Diagram," Revision 12

01-C-ZJS-591, "Control Building, Unit 2, Battery Rooms A, B, C, and D PK Battery Rack Arrangement," Revision 1

02-E-PKA-001, "Main Single Line Diagram, Class 1E 125V DC Class 1E and 120 V AC Inst Power System," Revision 5

02-M-HJP-002, "Control Building HVAC P&I Diagram "

02-C-ZJS-591, "Control Building, Unit 2, Battery Rooms A, B, C, and D PK Battery Rack Arrangement," Revision 0

03-E-PKA-001, "Main Single Line Diagram, Class 1E 125V DC Class 1E and 120 V AC Inst Power System," Revision 4

03-M-HJP-002, "Control Building HVAC P&I Diagram"

03-C-ZJS-591, "Control Building, Unit 3, Battery Rooms A, B, C, and D PK Battery Rack Arrangement," Revision 0

Miscellaneous

13-E050B-24-3, "Installation, Operation and Maintenance Manual for Class 1E Batteries and Racks"

Engineering Document Change 2003-00115

Printout on Corrective Maintenance WOs dated January 27, 2004

PK System Design Basis Manual

Section 1R06: Flood Protection

CRDR

2659731

Drawings

13-P-ZCE-102, "Containment Building - Level A Plumbing Plan Between EL 80'-0" & 100'-0"," Revision 13

13-P-ZAE-103, "Containment Building - Level 1 Plumbing Plan Between EL 100'-0" & 120'-0"," Revision 8

13-P-ZAE-200, "Auxiliary Building - Level D Plumbing Plan Between EL 40'-0" & 51'-6"," Revision 12

13-P-ZAE-201, "Auxiliary Building - Level C Plumbing Plan Between EL 51'-6" & 70'-0"," Revision 6

01-M-RDP-002, "P&I Diagram Radioactivie Waste Drain System," Revision 11

02-M-RDP-002, "P&I Diagram Radioactivie Waste Drain System," Revision 11

03-M-RDP-002, "P&I Diagram Radioactivie Waste Drain System," Revision 11

Miscellaneous

Calculation 13-MC-ZA-0808, "MSSS Flooding at Elevation 81'," Revision 4

Section 1R12: Maintenance Implementation

WOs

2414083, 2660243, and 2660246

CRDRs

320294 and 2660246

Procedures

32MT-9ZZ34, "Maintenance of Medium Voltage Circuit Breakers Type AM-4.16-250," Revision 13

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

CRDRs

2686271, 2686201, and 2686238

WOs

2669472

Section 1R15: Operability Evaluations

CRDRs

2596979, 2667729, 2578249, 2668992, 2664722, and 2634792

WM

2667723

Procedures

50AL-9DG02, "Diesel Generator B Alarm Panel Responses," Revision 14
40OP-9SI01, "Shutdown Cooling Initiation," Revision 30

Section 1R16: Operator Work-Arounds

Miscellaneous

Computer listing of operator workarounds as of January 14, 2004, and January 26, 2004

Section 1R19: Postmaintenance Testing

CRDR

2662111

Procedures

32ST-9ZZ34, "Battery Charger Surveillance Test," Revision 7
73ST-1XI12, "Safety Injection Train B ECCS Throttle Valves – Inservice Test," Revision 17

WOs

2586645, 2568455, 2571575, 2586472, 2586483, 2593557, 2630340, 2662112, 2662231, and 2683308

Permit

101574

Section 1R20: Outage Activities

Procedures

70DP-0RA01, "Shutdown Risk Assessment," Revision 8
40OP-9ZZ07, "Plant Shutdown Mode 1 to Mode 3," Revision 20
40OP-9ZZ16, "RCS Drain Operations," Revision 38
40OP-9ZZ20, "Reduced Inventory Operations," Revision 5
40OP-9ZZ11, "Mode Change Checklist," Revision 57
40OP-9ZZ03, "Reactor Startup," Revision 36

WOs

2685754, 2568616, 2568820, and 2568835

Miscellaneous

Licensing Document Change Request 04-R001, TRM T3.7.100 and associated Bases, "Steam Generator Pressure and Temperature Limitations"

Section 4OA3: Event Followup

CRDRs

CRDR 2592898, "(U-3) During the Performance of 73ST-9ZZ18, One of the MSSVs had an As-Found Setpoint Greater than Technical Specification Limit"

Significant CRDR Root Cause Investigation 2592898, "Unit 3 Main Steam Safety Valve (MSSV) High Test Results - MRFF Adverse ERCFA Investigation," Revision 0

CRDR 263684, "(U-2) CRDR Documents a Condition Where Non-seismic Tools Specifically Cranes May Have Been Used Over Operable Components"

Significant CRDR Root Cause Investigation 2636484, "Non-seismic Crane Used Over Operable Components During Refueling Outages"

Procedures

30DP-9MP12, "Overhead Hoisting Systems," Revision 11
31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revision 13
Design Basis Manual, "Category I Building Topical," Revision 4

Section 1EP2: Alert and Notification System Testing

Alert for Notification (Siren) Technical Description, Nuclear Operations Support Department, October 1982

"Emergency Broadcast Procedures for the Phoenix, Arizona, EBS Operational Area," May 1983

"APS Ambient Sound Level Measurement Study," Acoustic Technology Inc., October 1983

"Palo Verde Nuclear Generating Station Prompt Notification Siren System Test Report," October 1983

Standing Procedures for the Operation of the Notification and Alert Net, February 1984
Standing Procedures for the Operation of the Notification and Alert Net Backup, February 1984

Siren System Check Out Procedure, May 1984

"Palo Verde Nuclear Generating Station Site-Specific Offsite Radiological Emergency Preparedness Alert and Notification System Quality Assurance Verification," Federal Emergency Management Agency, August 1985

"Palo Verde Nuclear Generating Station Remote Control Siren System," Revision 3, May 2002

Special Assistance Request Survey [Card], Palo Verde Nuclear Generating Station, June 2002 Revision and January 2004 Revision

"Acoustic Analysis of the Siren Notification System for Palo Verde Nuclear Generating Station," Acoustic Technology, Inc., October 2002

Offsite Response Plan for Palo Verde Nuclear Generating Station, Annex E, "Warning," January 2001 Revision

Offsite Response Plan for Palo Verde Nuclear Generating Station, W&C, January 2002 Revision

System Description, E-Alert Receiver

Maricopa County Sheriff's Office Checklist(s) (Operations):
- Site Area Emergency
- General Emergency

Maricopa County Standard Operating Procedures for:
- Assistant Operations Officer
- Communications/Warning Officer
- Communications Shift Supervisor
- Scene Commander/Sheriff's Office

Palo Verde Siren System Procedures:
- "Activation of Siren System during Actual Emergency"
- "Activation of Siren Systems for Testing"
- "Silent Tests"
- "Siren Malfunction/Undesired Siren Activation"

<u>Section 1EP3: Emergency Response Organization Augmentation</u>

Guidance for Manual ERO Callout When Autodialer Not Available
DCC (Dialogics) Pre-Implementation Test Plan
Test Scenarios Response Drill, June 12, 2003
First Quarter 2003 DCC (Dialogics) test results, conducted January 29, 2004

<u>Section 1EP4: Emergency Action Level and Emergency Plan Changes</u>

EPIP-01, "On-shift Emergency Coordinator," Revision 14
EPIP-08, "Emergency Planning Administration," Revision 12

EPIP-99, Appendix H, Section 1.0, Revision 0

EPIP-99, "Standard Appendices," Form EP-0744, "Quarterly Pager/Autodialer Test Job Qualification Cards for Security Directors"

EPIP-99, "Standard Appendices, Form EP-0760, "10 CFR 50.54(Q) Screening Form"

EPIP-99, "Standard Appendices, Form EP-0761, "10 CFR 50.54(Q) Evaluation Form"

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

60DP-0QQ19, "Internal Audits," Revision 11

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

EPIP-99, "Standard Appendices," Form EP-0800, "ERO Comment Form"

NEP01-00-020, Emergency Planning Emergency Response Organization, Site Medical/Onsite Medical Staff, Job Qualification Card

NEP01-00-003, Emergency Planning Emergency Response Organization, Security Director TSC, Job Qualification Card

Self Assessment Reports:

EP-02-05, Review of EP kits , April 5, 2002
EP-02-01, STARS EP Program Review, April 25, 2002
Review of Proposed Revision 26 to E-Plan (ISFSI EALs), April 26, 2002
EP-02-14, Review of EP Kits, July 26, 2002
EP-02-09, Review of Contaminated/Injury Response Program, November 8, 2002
EP-03-03, EP Performance indicator program (March to Sept. 2003), December 12, 2003
EP-03-06, Kit Inventories, December 21, 2003

Emergency Preparedness Drill Reports:

EP-02-06, Review of PAR/NAN Performance during May 8, 2002, Drill, November 8, 2002
EP-03-01, ERO Full-Scale Drill, February 5, 2003
EP-03-08, Review of Contaminated Injury Response Evaluated Drill, November 11, 2003
EP-03-09, Review of Annual Offsite Siren Warning Test, November 22, 2003

Quality Assurance Audits:

Review and Updating of the Emergency Plan

USAR Section 13.4.5 (Audit Program)

NAD Audit Plan 02-009 (2002 Emergency Planning 54t Audit)

Audit Checklist 02-009-03, Objective 1 (Interface with offsite agencies)

Audit Report 02-009 (Emergency Planning)

STARS Emergency Preparedness Program Round Robin Self-Assessment for PVNGS, April 2002

Memorandum, dated February 13, 2003, "PVNGS 2003 Nuclear Assurance Audit Schedule," Revision 0

NAD Top Ten Identified Issues, December 2003

PVNGS Emergency Plan Revision 28:

Summary of changes
Form EP-0760, 10 CFR 50.54Q Screening
Administrative / Technical Review Checklist
PVNGS Emergency Plan Revision 27, Table 1

CRDRs

2488514	2514262	2534063	2590316	2592902
2494536	2516022	2550829	2590556	2608254
2510075	2516075	2576251	2592808	2613374
2512115	2518815	2583666	2596866	2651122
2513210	2532233	2587600		

Root Cause Analysis, CRDR 2592808, "Classification of SAE During an Emergency Planning Exercise," July 2003

LIST OF ACRONYMS

AFW	auxiliary feedwater
CFR	*Code of Federal Regulations*
CRDR	condition report/disposition request
HPSI	high pressure safety injection
LER	licensee event report
NCV	non-cited violation
RCS	reactor coolant system
SRA	senior reactor analyst
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
WM	work mechanism
WO	work order