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

February 13, 2004

Greg R. Overbeck, Senior Vice
President, Nuclear
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P. O. Box 52034
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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INSPECTION REPORT 05000529/2003009

Dear Mr. Overbeck:

On December 31, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Palo Verde Nuclear Generating Station, Unit 2, facility. No inspection of Units 1 or 3 were performed under this report number. The enclosed report documents the inspection findings, which were discussed on January 21, 2004, with you and other members of your staff.

This 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel. This inspection covers steam generator replacement activities at the Palo Verde Nuclear Generating Station.

Based on the results of this inspection, two findings of significance were identified. One of the findings concerned a heavy load drop in containment, and was determined to involve a violation of NRC requirements. This finding has potential safety significance greater than very low significance. This finding did not present an immediate safety concern. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in Section 4OA7 of this report. If you contest this noncited violation, 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 2.790 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/

Jeffrey A. Clark, Chief
Project Branch D
Division of Reactor Projects

Docket: 50-529

License: NPF-51

Enclosure:
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w/Attachment: Supplement Information

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-529
License:	NPF-51
Report No:	05000529/2003009
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Unit 2
Location:	5951 S. Wintersburg Tonopah, Arizona
Dates:	June 1 through December 31, 2003
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Approved By:	Jeffrey A. Clark, Chief, Project Branch D Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000529/2003009; 6/1/03 - 12/31/03; Palo Verde Nuclear Generating Station, Unit 2; Integrated Resident and Regional Report of Steam Generator Replacement Activities.

This report covered a 7-month period special inspection by resident and regional inspectors. The inspection identified two findings. 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. Inspectors Identified Findings

Cornerstone: Barrier Integrity

- Green. Proposed postmodification testing to determine the new heat losses to ambient term used in reactor thermal power calculations was inappropriate because it would have resulted in a nonconservative bias. Changes to the reactor coolant system components and new insulation were expected to cause a change in heat lost from the reactor coolant system. The licensee's software for calculating reactor thermal power included a constant term used to account for the reactor power lost in this way. The licensee planned to determine the new heat loss term by measuring it with the plant shutdown at the no-load operating temperature, and then applying it to all power levels. The proposed test would measure a lower heat loss term than would be present at full load power and temperatures, introducing a nonconservative bias in the calculated reactor power. The licensee estimated that the bias was expected to be about 0.3 MWth (.01 percent power). Since the output of this calculation was used to calibrate nuclear instrument reactor power and turbine power instruments, this bias would have caused a similar effect in these instruments.

  The safety significance of the proposed testing being nonconservative was very low, since the licensee planned to account for this condition prior to the implementation of the plant changes. This issue affected the Barrier Integrity Cornerstone objective for design control in maintaining fuel integrity. It was more than minor because if left uncorrected, it would be more significant because the licensee could inadvertently operate Unit 2 above its maximum licensed power level.

Cornerstone: Initiating Events

- TBD. The inspectors identified a violation having potential safety significance greater than very low significance. The violation occurred when personnel failed to follow a maintenance procedure preceding a heavy load drop inside containment. The load was dropped in the vicinity of reactor coolant and shutdown cooling system piping.

This finding is unresolved pending completion of a significance determination. The finding was greater than minor because it affects the initiating events cornerstone and had an actual impact in that a heavy load was dropped which is a precursor to a significant event. The finding also was determined to have potential safety significance greater than very low significance because of the increased likelihood of a loss of reactor coolant system inventory since the load movement occurred in the vicinity of reactor coolant system piping.

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.

- 10 CFR Part 50, Appendix B, Criterion X, in part, requires a program for inspection of activities affecting quality shall be established and executed to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Contrary to this, on numerous occasions during the steam generator replacement outage, conditions were identified that did not conform with the implementing documents after inspection and acceptance by Bechtel Field Engineering and/or Quality Control had been performed. These installation nonconforming conditions were identified in Nonconformance Reports 24199-107, 24199-096, 24199-121, 24199-124, and Vendor Corrective Action Reports VC-BBP7-03-047 and VC-BBP7-03-055. Each of these individual equipment conditions were corrected following their identification. The licensee's nuclear assurance department performed walkdowns of steam generator modifications that were either complete, or nearing completion, in response to this adverse trend. Vendor Corrective Action Report VC-BBP7-03-055 was initiated as a result of these walkdowns, and identified the need to assess the extent of condition as to the acceptability of other similar installations, and develop corrective actions to preclude future problems with Bechtel's quality assurance inspection program. This finding is of very low safety significance because all of the nonconforming conditions were corrected prior to core reload and mode escalation.

REPORT DETAILS

Plant Status

Unit 2 began the inspection period at essentially full power and remained at this level until September 27, 2003, when the unit was shutdown to commence the Unit 2 Eleventh Refueling Outage 2R11. The unit returned to essentially full power on December 23, 2003, following completion of the refueling outage.

1. REACTOR SAFETY
   Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluation of 10 CFR 50.59 Safety Evaluations for Steam Generator Replacement (71111.02)

a. Inspection Scope

The inspectors reviewed evaluations of changes, tests, or experiments associated with the steam generator replacement modifications. The inspectors reviewed seven permanent plant modification packages (design change packages) and the associated 10 CFR 50.59 evaluations. The inspectors interviewed the cognizant design and system engineers for the identified evaluations to determine their understanding of the evaluation and their conclusions.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

This inspection focused specifically on the Unit 2 steam generator replacement activities (e.g., nondestructive examinations (NDE) and welding activities). However, credit will be taken for the Unit 2 inservice inspection baseline program.

1. Performance of NDE Activities Other than Steam Generator Tube Inspections

The procedure requires review of two or three types of NDE activities (volumetric, surface and visual). The inspectors reviewed multiple examples of volumetric and surface examinations (as noted below in discussions of ultrasonic and magnetic particle testing). During the review of each examination, the inspectors verified that the correct NDE procedures were used, examinations and conditions were as specified in the procedures, and test instrumentation or equipment was properly calibrated and within the allowable calibration period.

The inspectors observed one ultrasonic examination performed on Field Weld FW-3 for Steam Generator 1 downcomer feedwater line, and reviewed the completed documentation. The inspectors also observed five magnetic particle examinations for the Steam Generator 1 downcomer feedwater line.

The procedure requires that if the licensee had completed welding on the pressure boundary for Class 1 or 2 systems since the beginning of the previous outage, verification be performed for one to three welds (e.g., radiography) that acceptance and preservice examinations were done in accordance with ASME Code. The inspectors reviewed three radiographic films taken of the replacement steam generator field welds as listed below:

- FW-2 (reactor coolant system, Steam Generator 1, nozzle-to-elbow), Drawing FSK-M-008

- FW-2 and FW-10 (feedwater pipe, Steam Generator 1, pipe-to-elbow), Drawing FSK-M-83

The inspectors also reviewed the documentation and radiographic film to determine if the indications revealed by the examinations were compared against the ASME Code specified acceptance standards. This review was also performed to determine if the indications were properly dispositioned. The NDE certifications of those personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspectors.

The procedure requires the review of one or two examinations from the previous outage with recordable indications that have been accepted by the licensee for continued service. The inspectors did not identify any previous examinations with recordable indications that were accepted for continued service. All recordable indications from previous examinations were corrected.

The procedure requires review of one or two ASME Section XI Code repairs or replacements to verify that Code requirements are met. The inspectors selected the Unit 2 pressurizer half-sleeve nozzle repair/replacement activity for review. This review was documented in Inspection Report 05000528/2003005; 05000529/2003005; 05000530/2003005.

The inspection procedure did not specify sample sizes for welding activities. However, Inspection Procedure 50001, “Steam Generator Replacement Inspection,” used as guidance, requires review of welding and NDE activities. The inspectors reviewed welding and NDE activities. Specifically, the inspectors observed welding of field welds on the reactor coolant system for Steam Generators 1 and 2 as listed below. This review included welders and welding procedure qualifications.

- FW-1 (elbow to pipe)
- FW-3 and 5 (elbow to nozzle)

2. Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors did not observe any steam generator tube activities because the new steam generators were in the process of being installed during this outage. However, the inspectors reviewed the baseline eddy current examination of the new steam generator tubes. Specifically, the inspectors reviewed the preservice eddy current summary report and the supporting eddy current data for a sample of identified flaws.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed two risk evaluations and overall plant configuration control for selected activities to verify compliance with Procedure 30DP-9MT03, “Assessment and Management of Risk When Performing Maintenance in Modes 1 - 4,” Revision 8, and "U2R11 Shutdown Risk Assessment," Revision 0. 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.

- September 8, 2003, evaluated controls and plans to minimize any adverse impact on Units 1 and 3 and common systems

- September 25 through October 3, 2003, movement of main steam piping during steam generator replacement outage

b. Findings

No findings of significance were identified.

1R17 Evaluation of Permanent Plant Modifications for Steam Generator Replacement (71111.17)

a. Inspection Scope

The inspectors reviewed procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications for the steam generator replacement program. The inspectors verified that these changes did not adversely affect the design and licensing basis of the facility. The inspectors reviewed seven permanent plant modification packages (design change packages) and documentation, including review screens and safety evaluations, to verify that they were performed in accordance with regulatory requirements and plant procedures. Procedures and modifications reviewed are listed in the attachment to this report.

The inspectors interviewed the cognizant design and system engineers for the identified modifications to determine their understanding of the modification packages. The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed the corrective action documents identified in the attachment to this report and the subsequent corrective actions pertaining to licensee-identified problems and errors in the performance of permanent plant modifications.

b. Findings

Introduction. The inspectors identified a Green finding regarding a nonconservative bias to reactor power calculations.

Description. The inspectors reviewed the impact to reactor power calculations caused by changes to the plant by steam generator replacement and power up-rate. This included testing plans following modification.

In the licensee's software for calculating reactor thermal power, secondary plant thermal power was measured and compared to reactor thermal power through known relationships. This relationship comparison included a constant term used to represent thermal power created in the reactor which was lost from the reactor coolant system to ambient prior to making it to the secondary plant. Changes to the reactor coolant system components and associated insulation were expected to cause some change in this term. The licensee planned to determine the new reactor coolant system heat losses to ambient term by measuring it with the plant shutdown at the no-load operating temperature, and then applying it to calculations performed at all power levels.

The inspectors pointed out that the proposed test would measure a lower heat loss term than would be present at full power because full-load operating temperatures (Tave and Thot) would be higher. This would have the result of introducing a bias in the calculated reactor power, causing it to be lower than actual (nonconservative). The licensee estimated that the bias was not more that 10 percent of the heat-loss term, which equated to 0.3 MWth (.01 percent power). Since the output of this calculation was used to calibrate nuclear instrument reactor power and turbine power instruments, this bias would have caused a similar effect in these instruments.

The inspectors also identified that the proposed testing method was the same as the method used to determine the existing heat losses to ambient term being used in all three units. This meant that a nonconservative bias of similar magnitude had existed in the reactor thermal power calculations since startup testing. This was determined not to be an immediate safety concern because all three units were operating 1-2 percent below the maximum licensed power levels to reduce steam generator tube degradation. However, the possibility existed that each unit may have operated in excess of its maximum licensed power level at some time in the past. The licensee initiated Condition Report/Disposition Request 2631624 to evaluate reportability and possible corrective actions.

The inspectors examined the licensee's review of past operations to see if any of the three units had exceeded their maximum licensed power level. The licensee concluded that none of the units had exceeded 100 percent power for more than 8 hours, accounting for the nonconservative bias. The inspectors concluded that the licensee's method of performing this evaluation was reasonable.

Analysis. The safety significance of operating with a small nonconservative bias in the reactor thermal power calculation was very low. The licensee demonstrated that the estimated value of the bias was less than the available margin in this calculation. This meant that while it was possible that each unit had operated slightly above its maximum licensed power level, none would have operated above its maximum analyzed power level used in safety analyses. The safety significance of the proposed postmodification test plan including a test method that would have introduced a similar bias was also determined to be very low.

Enforcement. A (Green) finding was identified because the licensee planned to use a nonconservative method for calculating heat losses to ambient in the reactor thermal power calculation that would have been implemented for the new steam generator replacement. No violation of regulatory requirements occurred because the modifications had not been implemented. The licensee placed this issue in their corrective action program as Condition Report/Disposition Request 2631624 (FIN 05000529/2003009-01).

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors witnessed or reviewed the results of postmaintenance testing for the following four maintenance activities:

- December 11, 2003, Procedure 40TI-9ZZ04, "SGRP, AFW Waterhammer Test," Revision 0
- December 10-19, 2003, hot gap measurements per Document 24199-M-018, "Thermal Expansion Monitoring Program for Unit 2 Steam Generator Replacement," Revision 1
- December 11, 2003, Procedure 72TI-9RC02, "Reactor Coolant System Flow Verification Following Steam Generator Replacement," Revision 0

- December 16 and 17, 2003, Procedure 40TI-9ZZ01, "SGRP Control System Checkout Test," Revision 1

In each case, the test procedures were reviewed to determine if the test adequately verified proper performance of the components affected by outage maintenance activities. The Updated Final Safety Analysis Report, Technical Specifications, and design-basis documents were also reviewed as applicable to determine the adequacy of the acceptance criteria listed in the test procedures.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed plant conditions and observed selected refueling outage activities associated with the Unit 2 Eleventh Refueling Outage to verify that the licensee maintained the plant in a configuration consistent with the requirements of Technical Specifications and with the assumptions of the outage risk assessment. For this inspection, the inspectors reviewed the following activities as they related to entering conditions necessary for performing the steam generator replacement. Coverage of the full scope of Inspection Procedure 71111.20 is documented in Inspection Report 05000529/2003005. The inspectors observed portions of the following activities:

- Monitoring of reactor shutdown and plant cooldown activities
- Reduced inventory and midloop conditions
- Refueling activities
- Clearance activities
- Monitoring of heatup and startup activities

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors evaluated the adequacy of periodic testing of the following important nuclear plant equipment. This review included aspects such as preconditioning, the adequacy of acceptance criteria, test frequency, procedure adherence, record keeping, the restoration of standby equipment, the effectiveness of the licensee's problem identification and resolution program, and test equipment accuracy, range, and calibration. The inspectors reviewed the following tests, which were performed after completing substantial replacement of reactor coolant system pressure boundary components:

- December 19, 2003, Procedure 40TI-9ZZ03, "SGRP Unit Load Transient Test," Revision 0

- December 17, 2003, Procedure 72PY-9RX04, "Low Power Physics Test Using RMAS," Revision 5

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following four temporary modifications with respect to design bases, approvals, and tracking. The inspectors reviewed the associated 10 CFR 50.59 screening, updated procedures, and drawings. The inspectors also walked down the temporary modification.

- 221244, "Containment Structural Modifications," Revision 0

- 221896, "Temporary Structures and Equipment Inside Containment," Revision 0

- 2404038, "Containment Penetration 58 Temporary Modification," Revision 1

- 2414057, "Steam Generator Wall Removal & Restoration and Reactor Cavity Wall," Revision 0

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (50001, 71121.01)

a. Inspection Scope

The inspector reviewed work activities involving radiological controls for airborne radioactivity areas, radiation areas, and high radiation areas for the Unit 2 steam generator replacement work. The following items were reviewed and compared with regulatory requirements:

- Area postings and other access controls for steam generator work activities

- Audits and self-assessments involving high radiation area controls and staff performance (Nuclear Assurance Evaluation Reports ER 02-0146 and ER 02-0151)

- Steam generator radiation exposure permits and associated radiological surveys that involved potential airborne radioactivity areas and high radiation areas (Radiation Exposure Permits: 2-6006A, SGRP Primary Side (RCS) Work; 2-6009A, SGRP Decon Activities; and 2-6010A, SGRP Pipe End Decon)

- Contamination control activities

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector reviewed work activities involving radiological controls for airborne radioactivity areas, radiation areas, and high radiation areas. The inspector interviewed radiation protection staff members and other radiation workers to determine the level of planning, communication, ALARA practices, and supervisory oversight integrated into work planning and work activities for the Unit 2 steam generator replacement work. In addition, the following items were reviewed and compared with procedural and regulatory requirements:

- Two ALARA prejob, in progress, and postjob reviews and associated radiation exposure permit packages from Refueling Outage 2R11 steam generator replacement activities which resulted in some of the highest personnel collective exposures

- ALARA work activities evaluations, exposure estimates, and exposure mitigation requirements

- Work activity intended dose against actual dose received and the reasons for any inconsistencies

- Method for adjusting exposure estimates, or replanning work, when unexpected changes in job scope or emergent work were encountered

- Use of engineering controls to achieve dose reductions and the benefits afforded by using shielding

The inspector completed five supplemental inspection requirements.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

1. Welding and NDE Inspection (71111.08)

a. Inspection Scope

The inspectors reviewed inservice inspection-related condition reports issued during the past year and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

The inspectors reviewed the corrective action documents issued during the current outage and reviewed in detail a sample of four condition reports on the steam generator welding and nondestructive testing activities. The inspectors verified that the licensee identified, evaluated, corrected, and trended in accordance with the program requirements in place at the Palo Verde Nuclear Generating Station.

b. Findings

No findings of significance were identified.

2. Steam Generator Replacement Outage Inspection (50001)

a. Inspection Scope

The inspectors reviewed the daily condition report summaries and nonconformance reports issued during the replacement project for risk-significant issues to see that the licensee was properly implementing the corrective action program. The inspectors verified that the licensee identified, evaluated, corrected, and trended in accordance with the program requirements in place at the Palo Verde Nuclear Generating Station. The inspectors also reviewed the licensee's actions to identify and correct lessons learned from the Unit 2 steam generator replacement project.

b. Findings

No findings of significance were identified.

3. Problem Identification and Resolution Process Review (71121.02)

a. Inspection Scope

Section 2OS2 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices.

b. Findings

No findings of significance were identified.

4. Heavy Load Drop Significant Investigation Review (50001)

a. Inspection Scope

The inspectors selected Significant CRDR 2639721, "Unit 2 Steam Generator Support Lever Heavy Load Drop," Revision 0, for detailed review. The inspector's assessment of this event is documented in Section 4OA5. The report was reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified, prioritized, and implemented. The inspectors evaluated this CRDR against the requirements of Procedure 90DP-0IP10, "Condition Reporting," Revision 16, and 10 CFR Part 50, Appendix B.

b. Observations and Findings

There were no findings identified associated with the root cause analysis, and corrective actions specified and implemented. The inspectors observed that immediate corrective action taken by the licensee was to suspend all heavy load lifts in containment until the completion of core off-load and until adequate review could be accomplished to identify and correct the cause of this event. Further, the inspectors observed that adequate corrective actions were implemented for subsequent heavy load lifts to preclude repetition of the event, therefore, no violation of regulatory requirements or findings were identified.

4OA5 Steam Generator Replacement Activities (50001)

1. Design and Planning Inspections

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 50001 to perform the following steam generator design and planning inspection activities.

Engineering and Technical Support

Inspections to review engineering and technical support activities were performed prior to, and during, the steam generator replacement outage by resident and regional office-based specialist inspectors. The results of the inspection are documented in Sections 1R02, 1R17, and 1R23.

Lifting and Rigging

Inspections to review engineering design, modification, and analysis associated with steam generator lifting and rigging activities were performed by resident and regional inspectors.

Security Considerations and Adverse Impact to Other Unit

Inspectors checked for potential adverse impacts to Units 1 and 3 (the nonoutage units) caused by outage activities, equipment configurations, etc., in accordance with Inspection Procedure 50001. The inspectors made frequent observations of security practices to verify that the licensee provided appropriate support for affected vital and protected area barriers during outage activities.

b. Findings

No findings of significance were identified.

2. Steam Generator Removal and Replacement Inspections

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 50001 to perform the following steam generator removal and replacement inspection activities.

Welding and NDE Activities

An inspection to review welding and NDE activities was performed during the steam generator replacement outage by regional office-based specialist inspectors. The results of the inspection are documented in Section 1R08.

Lifting and Rigging Activities (50001 and 71111.23)

The inspectors observed and reviewed several activities throughout the outage associated with lifting and rigging. The inspectors observed and reviewed preparations, procedures, crane and rigging inspections, and lay-down areas associated with the following activities:

- Construction of the outside lift system

- Partial bio-wall and interference removal and replacement
- Temporary handling device construction and removal
- Reactor cavity decking construction and removal
- Old steam generator removal
- Onload of new steam generator
- Transport of old steam generator to storage facility

Major Structural Modifications

The inspectors observed the implementation and reviewed documentation related to several structural modifications. The inspectors observed and reviewed the removal and reinstallation of the following structural modifications to support removal and replacement of steam generators (Section 1R23):

- Containment bio-wall removal as interference

- Structural supports for steam generator and all attached piping during all phases of removal and installation of the steam generator

Containment Access and Integrity

This was not applicable to Palo Verde Nuclear Generating Station steam generator replacement. The cutting of the outer containment wall was not necessary.

Outage Operating Conditions

The inspectors used Inspection Procedure 71111.20 to verify proper outage conditions. Section 1R20 records the activities reviewed.

Radiation Protection Controls

An inspection to review radiation protection controls was performed during the steam generator replacement outage by regional office-based specialist inspectors. The results of the inspection are documented in Sections 2OS1 and 2OS2.

Foreign Materials Control

The inspectors performed frequent observations of the steam generator replacement activities to verify the licensee was implementing proper foreign materials controls. In particular, the inspectors observed controls related to reactor coolant system and secondary side openings.

Temporary Services

The inspectors reviewed the work package and drawings, then observed the installation, use, and removal of temporary services in the containment building during the outage.

Instructions for the use and controls for construction power, acetylene, oxygen, and argon were reviewed, and the actual installation of each system was compared to the approved system sketches.

Storage of Old Steam Generator

The inspectors observed the transport and storage of the old steam generator to the onsite storage facility. The radiological safety plans were reviewed.

b. Findings

Introduction. A finding was identified for failing to follow a maintenance procedure preceding a heavy load drop inside containment. This is an unresolved item (URI) pending completion of the significance determination process.

Description. On October 3, 2003, a rigging operation to remove a 7000-lb. steam generator snubber lever plate was being performed by Bechtel Construction using the polar crane auxiliary hoist system. The plant was in Mode 6 with fuel in the core, the fuel pool was filled, and core off-load had not yet commenced.

The snubber lever became bound in the wall-mounted bracket during the rigging operation. The rigging crew tried to lower the auxiliary hoist, but it would not lower due to a known deficiency in that the mechanical load brake on the auxiliary hoist intermittently engaged. The work-around to disengage the mechanical brake was to raise the hoist a small amount. Two small pulls were attempted but the brake did not disengage. Bechtel supervision checked the condition of the crane cables and determined that the hoist could not be raised or moved any further. They pursued a plan to cut the steam generator arm interference. In preparations for the cutting evolution, an ironworker requested the Signalman to trolley the crane to move the lever before he started to cut the arm. When the crane trolleyed, it exerted additional strain on the cable and swivel hoist ring connected to the lever causing the swivel hoist ring to fail. The lever dropped 12-24 inches, coming to rest on the snubber and associated grating.

The licensee’s investigation determined that the event was the result of a series of errors. These errors included two root causes: inadequate Signalman self-checking with the person-in-charge before trolleying the crane; and inadequate communications regarding the work-around condition of the crane to Bechtel Operating Engineers and rigging crews. Contributing causes identified included poor verbal communications, prejob brief, and equipment performance. Additionally, numerous error precursors were identified, which potentially provoked the errors and/or inhibited the defenses against mitigating this event. The problem identification and resolution aspects of this issue are documented in Section 4OA2.4.

Analysis. The deficiency associated with this event was the failure to follow Procedure 31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revision 8, which led to the breakdown in numerous barriers which should have been in place to prevent the load drop. The finding was greater than minor because it affects the initiating events cornerstone and had an actual impact in that a heavy load was dropped which is a precursor to a significant event. The finding also was determined to have potential safety significance greater than very low significance because of the increased likelihood of a loss of reactor coolant system (RCS) inventory since the load movement occurred in the vicinity of RCS piping.

Enforcement. Technical Spceification 5.4.1 states, "Written procedures shall be established, implemented, and maintained covering the following activities: (a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; . . . ." Regulatory Guide 1.33 directs the licensee to establish and implement procedures for performing maintenance. Contrary to the above, on October 3, 2003, maintenance personnel failed to conduct a prejob sensitive issues briefing as required by Procedure 31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revision 8, prior to the movement of the snubber lever plate. Failing to perform the sensitive issues briefing had a significant impact on the focus of personnel involved in this critical maintenance evolution, contributing to the series of errors described above which led to the event. Pending determination of the finding's safety significance, this finding is identified as URI 05000529/2003009-01, "Failure to Follow Heavy Load Movement Procedure."

3. Postinstallation Verification and Testing Inspection

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 50001 to perform the following postinstallation verification and testing inspection activities.

Containment Testing

This was not applicable to Palo Verde Nuclear Station. The cutting of the outer containment wall was not necessary.

Licensee's Postinstallation Inspections and Verifications

The inspectors observed the implementation and reviewed several postinstallation surveillance and tests conduced under the licensee's return to service program. Specific items reviewed are documented in Sections 1R19 and 1R22. Specifically, the inspectors reviewed changes to the licensee's program for reactor coolant system and secondary side leakage testing due to the newly installed steam generator. The inspectors also reviewed the response of steam generator level and flow system controls after the licensee recalibrated the instrumentation affected by steam generator replacement.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

On January 21, 2004, the team presented the inspection results to Mr. David Mauldin, Vice President, Engineering and Support, and other members of licensee management. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was identified.

The inspectors presented the permanent plant modifications and evaluation of changes, tests, or experiments inspection results to Mr. Carl Churchman, Director, Steam Generator Replacement Project, and other members of licensee management at the conclusion of the inspection on August 29, 2003. The licensee acknowledged the findings presented.

On October 10, 2003, the inspector presented the inspection results to Mr. G. Overbeck, Senior Vice-President, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

The inspectors presented the results of the inservice inspection effort to Mr. Mike Winsor, Director, Engineering, and other members of licensee management at the conclusion of the inspection on October 24, 2003, and with Mr. David Mauldin, Vice President, Engineering and Support on November 7, 2003. The licensee acknowledged the findings presented.

On December 19, 2003, the inspector presented the inspection results to Mr. J. Gaffney, Director, Radiation Protection, and other members of your staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- 10 CFR Part 50, Appendix B, Criterion X, in part, requires a program for inspection of activities affecting quality shall be established and executed to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Contrary to this, on numerous occasions during the steam generator replacement outage, conditions were identified that did not conform with the implementing documents after inspection and acceptance by

Bechtel Field Engineering and/or quality control had been performed. These installation nonconforming conditions were identified in Nonconformance Reports 24199-107, 24199-096, 24199-121, 24199-124, and Vendor Corrective Action Reports VC-BBP7-03-047 and VC-BBP7-03-055. Each of these individual equipment conditions were corrected following their identification. The licensee's nuclear assurance department performed walkdowns of steam generator modifications that were either complete, or nearing completion, in response to this adverse trend. Vendor Corrective Action Report VC-BBP7-03-055 was initiated as a result of these walkdowns, and identified the need to assess the extent of condition as to the acceptability of other similar installations, and develop corrective actions to preclude future problems with Bechtel's quality assurance inspection program. This finding is of very low safety significance because all of the nonconforming conditions were corrected prior to core reload and mode escalation.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

C. Churchman, Director, Steam Generator Replacement Project
K. Coon, Technical Management Assistant, Radiation Protection
C. Corcoran, Senior Engineer, Steam Generator Replacement Project
D. Coxon, Shift Manager, Nuclear Operations
M. Fladager, Department Leader, Radiation Protection
J. Gaffney, Director, Radiation Protection
T. Gray, Department Leader, Radiation Protection
K. Greenspan, Senior Project Manager, Steam Generator Replacement Project
M. Karbassian, Section Lead, Steam Generator Replacement Project
D. Marks, Section Leader, Regulatory Affairs
G. Overbeck, Senior Vice-President
M. Pacholke, Senior Project Manager, Steam Generator Replacement Project
S. Peace, Consultant, Communications
R. Pontes, Section Lead, Steam Generator Replacement Project
R. Prabhakar, Senior Project Quality Manager, Steam Generator Replacement Project
M. Shea, Director, Nuclear Training
T. Weber, Section Leader, Regulatory Affairs
D. Wheeler, Section Leader, Nuclear Assurance-Engineering

Others

J. Bayless, Inservice Inspection Engineer
S. Bauer, Department Lead, Regulatory Affairs
F. Gowers, Site Representative, El Paso Electric
D. Hanson, Inservice Inspection Engineer
D. Hautala, Licensing Engineer
R. Henry, Site Representative, Public Service of New Mexico
R. Indap, Inservice Inspection Engineer
D. Marks, Section Leader, Regulatory Affairs
D. Mauldin, Vice President, Engineering and Support
F. McDougall, Contract Services Project Manager, Bechtel
E. McGilley, Project Quality Manager, Bechtel
M. Melton, Section Lead, Inservice Inspection Engineer
G. Michael, Regulatory Affairs
M. Powell, Department Lead, Maintenance Engineering
M. Sontag, Department Lead, Nuclear Assurance
J. Taylor, Contract Services, Executive Oversight
M. Winsor, Director of Engineering

NRC

N. Salgado, Senior Resident Inspector
G. Warnick, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000529/2003009-01	FIN	Failure to use a conservative method to calculate reactor coolant system heat losses for postmodification testing (Section 1R17)
05000529/2003009-01	URI	Failure to follow heavy load movement procedure (Section 4OA5)

Closed

05000529/2003009-01	FIN	Failure to use a conservative method to calculate reactor coolant system heat losses for postmodification testing (Section 1R17)

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 1R02: Evaluation of 10 CFR 50.59 Safety Evaluations for Steam Generator Replacement

Procedures

93DP-0LC07, "10CFR50.59 and 72.48 Screenings and Evaluations," Revision 6

Section 1R08: Inservice Inspection Activities

Procedures

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

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

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

73TI-9ZZ05, "Dry Magnetic Particle Examination," Revision 10

73TI-9ZZ06, "Wet Magnetic Particle Examination," Revision 11

73TI-0ZZ13, "Radiographic Examination," Revision 9

73TI-0EE01, "Ultrasonic Instrument Calibration," Revision 3

30DP-0WM12, "Housekeeping and System Cleanliness," Revision 10

90DP-0IP10, "Condition Reporting," Revision 16

Work Orders

2513813	2385021
2643126	2561913
2614243	2316467

Condition Report/Disposition Request

2579575	2601733
2579129	2626419
2581487	2643882
2580187	

Miscellaneous Reports

PV04Q401, "Design Report, Palo Verde Nuclear Generating Station Units 1, 2, and 3 Pressurizer Heater Sleeve Outside Diameter Weld Repair," Revision 0

Qualification Records Reviewed

Lambert MacGill & Thomas -Section XI, incl. pressurizer nozzle repair/replacement: Reviewed qualification records for 3 Level II and 3 Level III inspectors - all PDI qualified

Bechtel - only steam generator replacement - Reviewed qualification records for 2 Level III inspectors - no PDI qualified

MQS - only steam generator replacement Inspections: Reviewed qualification records for 16 Level II inspectors - no PDI qualified

Drawing

02-P-SGR-156, Downcomer Feedwater Line for Steam Generator 1 Field Welds 2, 3, 4, 5, and 6

UT Examination/Calibration Reports

UT-03-177	UT-03-181
UT-03-178	UT-03-182
UT-03-179	

MT Examination Reports

MT-03-123	MT-03-126

MT-03-124	MT-03-127
MT-03-125	

Radiographic Test Report

RT-03-029	RT-03-019
RT-03-029/059	RT-03-186
RT-03-028	RT-03-027
RE-03-069/028	

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Condition Report/Disposition Request

97Q525, Action 2

Procedures

40OP-9ZZ16, "RCS Drain Operations," Revision 33
31MT-9ZC07, "Miscellaneous Containment Building Heavy Loads," Revisions 8 and 9
72IC-9RX03, "Core Reloading," Revision 22

Miscellaneous

10 CFR 50.59, "Procedure 31MT-9ZC07, Steam Generator Replacement Project Main Steam Pipe Spool Lifts," Revision 0

Updated Final Safety Analysis Report, Section 9.1, "Fuel Storage and Handling," Revision 12

NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment"

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"

Engineering Study 02-MS-B033, "Unit 2 Steam Generator Replacement Project Main Steam Pipe Spool Lifts Evaluation," Revison 0

445-0315-GRD, "U2R11 Shutdown Risk Assessment," Revision 0

Section 1R17: Evaluation of Permanent Plant Modifications for Steam Generator Replacement

Procedures

81DP-0EE10, “ Plant Modifications,” Revision 10

81DP-0DC13, “Deficiency Work Order,” Revision 14

12DP-0MC29, “Warehouse Discrepancy Notice,” Revision 13

90DP-0IP10, “Condition Reporting,” Revision 16

P43-T-o (82), “Bechtel Welding Procedure Specification,” Revision 2

P43-T-Ag, “Bechtel Welding Procedure Specification,” Revision 3

40DP-9AP20, “Steam Generator Replacement and Power Uprate Startup Test Program,” Revision 0

81DP-0EE10, “Plant Modifications,” Revision 10

TA-13-C00-2001-014, “PVNGS Secondary Calorimetric Power Uncertainty Analysis,” Revision 1

72TI-9RC03, Reactor Coolant System Heat Loss Measurement Following Steam Generator Replacement,” Revision 0

40TI-9ZZ01, “SGRP Control Systems Checkout Test,” Revision 0

40TI-9ZZ03, SGRP Unit Load Transient Test,” Revision 0

Design Master Work Order (DMWO)

DWMO 2374890, “Convert Unit 2 Main Turbine from Partial to Full Arc,” Revision 1

DMWO 2345105, ”Modify 2JSGEUV0169 and 2JSGEUV0183 to Increase Actuator Thrust Capability,” Revision 0

DMWO 12-221893, Revision 1, (P3 package: RSG Preparation)

DMWO 13-221242, Revision 1, Steam Generator Large Bore Piping (S3 package)

DMWO 2417485, “NSSS Control Systems Modifications to Support Power Uprate/Steam Generator Replacement,” Revision 0

DMWO 2412303, “Instrumentation Calibration Change Implementation,” Revision 0

DMWO 222063, "Spray Pond Temperature Instrumentation Replacement," Revision 1

50.59 Evaluations

S-02-0406, “DMWO 2412303 - Calibration of Selected Instruments Affected by SGR/PUR to New Values,” Revision 0

S-02-0419, “DMWO 2417485 - Control System Modifications,” Revision 1

S-02-0372, DMWO 12-221893, Revision 1

S-02-0405/E-02-0034, DMWO 13-221242, Revision 1

S-02-0407, DMWO 2374890, "Convert Unit 2 Main Turbine from Partial to Full Arc," Revision 1

S-03-0009, DMWO 2345105, "Modify 2JSGEUV0169 and 2JSGEUV0183 to Increase Actuator Thrust Capability," Revision 0

S-01-00012, DMWO 222063, "Spray Pond Temperature Instrumentation Replacement," Revision 1

Condition Report/Disposition Request

2436467	2588675
2577331	2588676
2583674	2589740
2583730	2619106

Calculations

13-MC-MT-200, "Secondary Plant Thermal Parameters For 102% and 105% Power Uprate," Revision 4

13-MC-ZZ-219, "AOV Thrust and Actuator Sizing – Anchor/Darling Gate Valves," Revision 3

R96.081, "Maximum Thrust Analysis Report 4 inch Class 900 Carbon Steel Flex Wedge Gate Valve with 14 inch Bore Fail Close Air Actuator," Revision A

72TI-9RC02, "Reactor Coolant System Flow Verification Following Steam Generator Replacement," Revision 0

V-PENG-CALC-007, Natural Circulation Cooldown Analysis at Uprated Power Conditions," Revision 1

Engineering Study 02-JS-A086, "Power Uprate Steam Generator Replacement Project NSSS Control System Evaluation"

Miscellaneous

10106-112, "Sargent and Lundy Evaluation for the effects of U2 power uprate on plant equipment," Revision 0

Unit 2 Loose Part List

ANSALDO Nonconformity Report GNC 3220

Section 1R19: Postmaintenance Testing

Reports

BE-APS-2003-125, "Hot Gap Report - 350 degree and 560 Degree Plateaus," December 12, 2003

Nonconformance Report (NCR) 24199-135

Condition Report/Disposition Request

2657004 2657008

Section 1R20: Refueling and Outage Activities

Permits

94039 93487

Section 1R23: Temporary Plant Modifications

Calculations

02-CC-ZC-0195, "Miscellaneous Steel Platforms and Walkways," Revision 12

13-CC-ZC-0285, "Heat Sink Calculations for Containment Building," Revision 25

24199-C-020, "Seismic II/I Evaluation of Partial Removal of North D-Ring and Refueling Canal Wall," Revision 25

02-CC-ZC-0397, Auxiliary Crane Support Tower," Revision 0

24199-C-003, "Evaluation of Material Handling System Inside Containment," Revision 0

24199-C-011, "Auxiliary Crane #1 Support Frame," Revision 0

Section 2OS2: Access Control to Radiologically Significant Areas

Radiation Exposure Permits

2-6002 Steam Generator Replacement Support Activities
2-6006 Steam Generator Replacement Primary Piping (RCS) Work

Quality Assurance Audits and Surveillances

Nuclear Assurance Evaluation Report ER 02-0256
Nuclear Assurance Evaluation Report ER 03-0102

Section 4OA2: Identification and Resolution of Problems

Bechtel Nonconformance Report 24199-115, “Foreign Materials in (feedwater) Pipe”

Section 4OA5: Steam Generator Replacement Activities

Temporary Services

Temporary Module Package 2404038, “Containment Penetration Temporary Modifications”

Temporary Module Package 2373777, “13.8 KV Transformer and Temporary Power for Containment”

Storage of Old Steam Generator

Work Order 221246, “Unit 2 Old Steam Generator Storage Facility”

Procedures

30DP-0WM12, "Housekeeping and System Cleanliness," Revision 10
30DP-9MP01, "Conduct of Maintenance," Revision 35

Licensee’s Postinstallation Inspections and Verifications

Procedure 40ST-9RC02, "ERFDADS (Preferred) Calculation of RCS Water Inventory," Revision 24

4OA7: Licensee-Identified Violations

Procedures

"Installation Oversight Plan," Revision 2
CP-07, "Construction Procedure," Revision 2
"Nuclear Quality Assurance Manual," Revision 4
"Nuclear Quality Control Manual," 1997 Edition