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

January 30, 2004

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

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000528/2003005, 05000529/2003005, AND 05000530/2003005

Dear Mr. Overbeck:

On December 31, 2003, the US 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 January 7, 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 two inspector identified findings of very low safety significance (Green) and one unresolved item pending significance determination. 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 two 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 2.790 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/

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

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

Enclosure:
NRC Inspection Report 05000528/2003005, 05000529/2003005, and 05000530/2003005
w/attachment: Supplement Information

cc w/enclosure:
Steve Olea
Arizona Corporation Commission
1200 W. Washington Street
Phoenix, AZ 85007

Douglas K. Porter, Senior Counsel
Southern California Edison Company
Law Department, Generation Resources
P.O. Box 800
Rosemead, CA 91770

Chairman
Maricopa County Board of Supervisors
301 W. Jefferson, 10th Floor
Phoenix, AZ 85003

Aubrey V. Godwin, Director
Arizona Radiation Regulatory Agency
4814 South 40 Street
Phoenix, AZ 85040

Craig K. Seaman, Director
Regulatory Affairs/Nuclear Assurance
Palo Verde Nuclear Generating Station
Mail Station 7636
P.O. Box 52034
Phoenix, AZ 85072-2034

Hector R. Puente
Vice President, Power Generation
El Paso Electric Company
340 E. Palm Lane, Suite 310
Phoenix, AZ 85004

Terry Bassham, Esq.
General Counsel
El Paso Electric Company
123 W. Mills
El Paso, TX 79901

John W. Schumann
Los Angeles Department of Water & Power
Southern California Public Power Authority
P.O. Box 51111, Room 1255-C
Los Angeles, CA 90051-0100

John Taylor
Public Service Company of New Mexico
2401 Aztec NE, MS Z110
Albuquerque, NM 87107-4224

Cheryl Adams
Southern California Edison Company
5000 Pacific Coast Hwy. Bldg. DIN
San Clemente, CA 92672

Robert Henry
Salt River Project
6504 East Thomas Road
Scottsdale, AZ 85251

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, TX 78701-3326

Electronic distribution by RIV:
Acting Regional Administrator **(TPG)**
DRP Director **(ATH)**
Acting DRS Director **(ATG)**
Senior Resident Inspector **(NLS)**
Branch Chief, DRP/D **(LJS)**
Senior Project Engineer, DRP/D **(JAC)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Mel Fields **(MBF1)**
PV Site Secretary **(vacant)**
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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:	05000528/2003005, 05000529/2003005, and 05000530368/2003005
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Tonopah, Arizona
Dates:	September 21 through December 31, 2003
Inspectors:	N. Salgado, Senior Resident Inspector, Project Branch D D. Carter, Health Physics Inspector, Plant Support Branch P. Goldberg, Reactor Inspector, Plant Engineering Branch M. Hay, Senior Resident Inspector, Waterford, Project Branch E C. Johnson, Senior Reactor Inspector, Engineering Branch B. McNeill, Reactor Inspector, Engineering and Maintenance Branch J. Melfi, Resident Inspector, Project Branch D J. Taylor, Reactor Inspector, Plant Engineering Branch W. Sifre, Reactor Inspector, Engineering Branch G. Warnick, Resident Inspector, Project Branch D G. Werner, Senior Operations Engineer, Operations Branch
Accompanying Persons:	J. Adams, Reactor Inspector, Engineering and Maintenance Branch G. George, Reactor Inspector Intern, Division of Reactor Safety
Approved By:	Jeffery A. Clark, Chief, Project Branch D Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000528/2003005, 05000529/2003005; 05000530/2003005; 9/21/03 - 12/31/03; Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Surveillance Testing, Identification and Resolution of Problems, Temporary Instruction 2515/152

This report covered a 3-month period of inspection by resident inspectors, the Waterford senior resident inspector, a health physics inspector, a senior reactor inspector, four reactor inspectors, and a senior operations engineer. Two Green noncited violations and two unresolved items were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management's review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

TBD. An apparent self-revealing violation of 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawing" was identified. Specifically, the failure to secure the Unit 2 main steam line knee brace for Pipe Whip Restraint Hanger 02-SG-042-H-890 in accordance with Drawings 13-C-ZCS-541, Section B, and 13-C-ZCS-542, Section D, was an apparent violation of 10 CFR Part 50, Criterion V. The licensee documented this deficiency in their corrective action process as Condition Report/Disposition Request 2643347. This finding does not represent an immediate safety concern because new anchor bolt nuts were installed during reassembly of the system as part of the Unit 2 steam generator replacement outage.

This finding is unresolved pending completion of a significance determination. The inspectors determined that this performance deficiency was more than minor because it affected the barrier integrity cornerstone to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The deficiency was determined to result in a potential open pathway affecting the physical integrity of reactor containment requiring use of Inspection Manual chapter 0609, Appendix H, "Containment Integrity." The inspectors concluded that the containment integrity function would only be affected during a postulated main steam line break scenario and that the exposure time for this degraded condition, coupled with the probability of core damage, would require further review in order to determine the risk significance of the issue.

Cornerstone: Mitigating Systems

Green. The inspectors identified a noncited violation for the licensee's failure to implement Surveillance Requirement 3.5.3.8 for all three units. The licensee failed to identify and remove debris in Trains A and B emergency core cooling system sumps during their last performance of Procedure 31ST-SI01, "Cleaning/Inspection of ECCS Sumps," Revision 7. Specifically, the licensee failed to identify unqualified tie-wraps that were attached to the stem of the containment sump suction valves inside the emergency core cooling system sumps.

This finding is greater than minor, since it affected the mitigating system cornerstone objective of equipment reliability because the debris could have affected containment spray pump flow by clogging spray nozzles. The finding is of very low safety significance because the amount of debris would have only degraded containment spray pump flow during a potential large break loss of coolant accident, but the safety function would have been fulfilled based on the small amount of debris.

Green. The inspectors identified a noncited violation related to 10 CFR Part 50, Criterion III, "Design Control." This violation is related to having an unscreened hole in each emergency core cooling system train's sump covers. These 1-inch holes were greater than the 1/8-inch gaps allowed by the emergency core cooling system sump design.

This finding is greater than minor because it affected the mitigating system cornerstone objective of equipment reliability by not assuring that the sump structure would filter out all debris greater than 3/16-inch diameter. The finding is of very low safety significance because the location of these holes were not in the design flowpath for water into the emergency core cooling system sump, which would have limited the amount of debris introduced into the system.

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially full power until October 15, 2003, when power was reduced to 75 percent due to an electrical protection trip on heater drain Pump B. Following repairs to the pump's motor electrical terminations, the unit was returned to essentially full power on October 17, 2003, and remained there for the duration of the inspection period.

Unit 2 operated at essentially full power until September 27, 2003, when the reactor was shut down for the eleventh refueling and steam generator replacement outage. The steam generator outage activities are included in NRC Inspection Report 05000529/2003009. On December 8, 2003, while in Mode 3, a plant cooldown to Mode 5 was required due to a secondary leak on steam Generator 1 downcomer feedwater header check Valve 2PSGEV652. The outage was completed on December 15, 2003. The unit was returned to essentially full power on December 23, 2003, and remained there for the duration of the inspection period.

Unit 3 operated at essentially full power until November 8, 2003, when power was reduced to 90 percent to implement condensate system repairs. The unit was returned to essentially full power on November 9, 2004, and remained there for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [REACTOR-R]

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 refueling water tank level transmitters would remain operable at temperatures below 40^{o}F. The inspectors also performed a walkdown of the area where the transmitters are located.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

1. Partial Walkdown

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

- On October 1, 2003, accessible portions of safety injection system Train A (Unit 2)
- October 28, 2003, emergency diesel generator (EDG) Train B (Unit 2)
- October 30, 2003, low pressure safety injection system Train B (Unit 1)
- November 19, 2003, steam generator piping associated with atmospheric dump Valve 179 (Unit 3)
- December 3, 2003, EDG Train A (Unit 1)

b. Findings

No findings of significance were identified

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted tours of six 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:

- October 15, 2003, containment building - steam generator replacement activities (Unit 2)
- October 21, 2003, auxiliary building 100-foot, 120-foot, 140-foot elevations (Unit 2)
- October 25, 2003, main steam support structure (Unit 2)
- October 25, 2003, containment building all elevations (Unit 2)
- October 25, 2003, condensate pump room and tunnels (Unit 1)
- October 25, 2003, condensate pump room and tunnels (Unit 3)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed Section 3.6.2 of the UFSAR and toured the facility to determine if the licensee had taken adequate precautions against internal flooding. The inspectors selected the EDG rooms and the auxiliary feedwater pump rooms that are risk significant and susceptible to flooding. In addition, the inspectors reviewed Calculations 13-MC-ZA-808, "MSSS Flooding at Elevation 81 Feet," Revision 2, and 13-MC-DG-201, "Diesel Generator Building Flooding Analysis," Revision 1. The inspectors performed a walkdown of the auxiliary feedwater pump rooms and the EDG rooms to determine if the conditions in the rooms matched the assumptions and data in the calculations to ensure that the internal flooding design basis would not be exceeded.

b. Findings

No finding of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

This portion of the inspection pertained to the pressurizer half sleeve nozzle repair and replacement activities for Unit 2. Additional inservice inspection activities were performed and documented in NRC Inspection Report 05000529/2003009.

Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

The procedure requires verification of one or two ASME Section XI Code repairs or replacements. The inspectors selected one activity for review, the Unit 2 pressurizer half-sleeve nozzle repair/replacement activity.

The inspectors reviewed the eddy current data regarding the inspection of the axial and circumferential flaws found in the pressurizer heater sleeves.

The inspectors observed a liquid penetrant examination performed on the inconel pad welds to the pressurizer.

The inspectors reviewed design documents and work packages in regard to the repair and replacement of the half sleeve nozzles.

The inspectors also reviewed ASME Code requirements and Code cases that the licensee had implemented in regard to the pressurizer half sleeve nozzle repair.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

Requalification Activities Review by Resident Staff

a. Inspection Scope

On November 6, 2003, the inspectors observed the operations crew performance during evaluated simulator Scenario SES-0-03-J-00, "Loss of Isophase Bus Cooling/LOOP/ISLOCA with Containment Isolation Failure," dated September 30, 2003. The inspectors evaluated the simulator scenario, the crew performance, and the evaluator critique sessions conducted following the completion of the simulator scenario.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11B)

1. Biennial Inspection

a. Inspection Scope

The inspector reviewed the annual operating examination test results for 2003. The last biennial written was administered in 2002 and was reviewed and documented in NRC Inspection Report 05000528/2002004; 05000529/2002004; 05000530/2002004. The 2003 annual operating examination test results were assessed to determine if they were consistent with NUREG 1021 guidance and Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process," requirements. This review included examination test results for 70 senior operators and 38 reactor operators.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

a. Inspection Scope

The inspectors verified the licensee’s appropriate handling of structure, system, and component performance or condition problems during review of the following two equipment failures. Additionally, the inspectors evaluated the following equipment failures to verify that licensee personnel properly implemented the requirements of

10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants":

- Historical indication failures related to excore Channel D described in CRDRs 2317434, 2508869, and 2440810 (Unit 2)

- October 17, 2003, motor-operated valve actuators identified with incorrect tripper finger assembly fasteners described in CRDR 2644780 (Unit 2)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Throughout this inspection period, the inspectors reviewed daily and weekly work schedules to determine when risk significant activities were scheduled. The inspectors reviewed risk evaluations and overall plant configuration control for six 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:

- October 10-11, 2003, replacement of EDG B motor-operated potentiometer per WO 2641476 (Unit 3)

- October 20, 2003, fuel sipping per WO 2632577 (Unit 2)

- December 3, 2003, scheduled online outage for EDG, essential spray pond, essential chilled water, essential cooling water, and containment spray Train A (Unit 1)

- December 9, 2003, corrective maintenance performed on Train B high pressure safety injection to Loop 2 check Valve 2PSIB-V532 per WO 2656827 (Unit 2)

- December 12, 2003, reactor vessel flange inner o-ring failure evaluated per CRDR 2658166 (Unit 2)

- December 17, 2003, scheduled online outage for EDG, essential spray pond, essential chilled water, essential cooling water, and containment spray Train A (Unit 3)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors observed the following nonroutine evolution to verify that it was conducted in accordance with license procedures and Technical Specification requirements:

- On December 4, 2003, the inspectors reviewed and observed the licensee's response to unexpected vibrations on the Unit 2 safety injection system following the start of Reactor Coolant Pumps 1A and 1B. The vibrations were noted on the lines upstream of the safety injection check valves to Cold Legs 1A and 1B and in the west piping penetration room. The licensee stopped the reactor coolant pumps and initiated CRDR 2656398 to evaluate this unexpected condition. The licensee implemented an action plan to attempt to recreate the vibration. The licensee verified that the affected safety injection lines were vented and filled the safety injection tanks. The licensee was unable to recreate the unexpected vibration.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- September 21, 2003, evaluation of Rosemount transmitter response to low ambient temperatures, documented in CRDRs 2585721, 2629698, and 2637745 (Unit 2)

- September 26, 2003, evaluation of the increasing trend in pressurization of the safety injection header documented in CRDR 2635507 (Unit 2)

- October 2, 2003, evaluation of the inoperable boron dilution monitor documented in CRDR 2639366 (Units 1, 2, and 3)

- October 22, 2003, Operability Determination 2644782, "Tripper Finger Bolt Issue for Limitorque SMB-0 Actuators," Revision 1 (Units 1, 2, and 3)

- October 24, 2003, Cylinder 9R air intake manifold offsticker leak on EDG 2B documented in deficiency WO 2645454 (Unit 2)

- December 4, 2003, evaluation of a tie-wrap remaining in guide tube for control element Assembly 80 documented in CRDR 2655298 (Unit 2)

- December 5, 2003, evaluation of the both containment sump covers having a 1-inch diameter hole documented in CRDR 2656229 (Unit 2)

- December 5, 2003, evaluation of lead shielding remaining installed on pressurizer surge line during heatup documented in CRDR 2656111 (Unit 2)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

1. Core Protection Calculators and Control Element Assembly Calculator (CPC/CEAC) Modification

a. Inspection Scope

The inspectors reviewed the Unit 2 CPC/CEAC Modification DMWO 223535 to verify that it was being performed in accordance with regulatory requirements and plant procedures. The inspectors interviewed the licensee personnel installing the modification as to their understanding of the modification package and observed work in progress. The inspectors also observed portions of the Unit 2 CPC/CEAC modification work to verify the following:

- work package was at work site
- transient combustible material was appropriately controlled
- construction material was appropriately staged
- construction debris was kept to a minimum

b. Findings

No findings of significance were identified.

2. High Pressure Safety Injection Modifications

a. Inspection Scope

The inspectors verified as-built configuration for two modifications on the Unit 3 safety injection system.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and/or evaluated the results from the following two 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.

- November 5, 2003, stroke of atmospheric dump Valve SGN-HV-179 with steam per Procedure 40OP-9SG01, "Main Steam," Revisions 31A and 31B, Appendix V, following troubleshooting activities (Unit 3)
- December 7, 2003, rework of the Train B high pressure safety injection to Loop 2 check Valve 2PSIB-V532 per WO 2656827 (Unit 2)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the licensee's Unit 2 11th refueling outage shutdown risk assessment to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Steam generator outage activities are included in NRC Inspection Report 05000529/2003009. Documents reviewed during the inspection are listed in the attachment.

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

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

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

- Monitoring of decay heat removal processes

- Ensuring that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system

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

- Activities that could affect reactivity

- Maintenance of secondary containment as required by Technical Specification

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage

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

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

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing (71111.22)</u>

a. <u>Inspection Scope</u>

The inspectors observed the performance of and/or reviewed documentation for the following surveillance tests. Applicable test data was reviewed to verify whether they met Technical Specification, UFSAR, and licensee procedure requirements. Also, the inspectors verified that the testing effectively demonstrated that the 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.

- September 29, 2003, Procedure 73ST-9DG02, "Class 1E Diesel Generator and Integrated Safeguards Test - Train B," Revision 7, Sections 8.1 and 8.2 (Unit 2)
- November 12, 2003, Procedure 73ST-9CL01, "Containment Leakage Type 'B' and 'C' Testing,"Revision 21, Section 8.19 (Unit 2)
- December 2, 2003, Procedure 31ST-9SI01, "Cleaning/Inspection of ECCS Sumps," Revision 7 (Unit 2 - Train A)
- December 3, 2003, Procedure 31ST-9SI01, "Cleaning/Inspection of ECCS Sumps," Revision 7 (Unit 2 - Train B)
- December 8, 2003, Procedure 73ST-9XI38, "AFA-PO1 Discharge Checkvalve AFA-V015 - Inservice Test," Revision 5 (Unit 2)
- December 31, 2003, Procedure 40ST-9DG02-3, "Diesel Generator B Test," Revision 23 (Unit 3)

b. Findings

1. Inspection of Emergency Core Cooling System (ECCS) Sumps

Introduction. The inspectors identified a noncited violation for the licensee's failure to implement Surveillance Requirement 3.5.3.8 for all three units. The licensee failed to identify and remove debris in Trains A and B ECCS sumps during their last performance of Procedure 31ST-9SI01, "Cleaning/Inspection of ECCS Sumps," Revision 7.

Description. On December 2-3, 2003, the inspectors observed the performance of Procedure 31ST-9SI01, which provides the requirements to verify cleanliness and sump integrity. The inspectors identified that a tie-wrap was used to secure a strain gauge onto each ECCS sump valve stem. The inspectors questioned whether the tie-wraps were qualified to remain intact during a postulated loss of coolant accident (LOCA). The licensee determined that these tie-wraps were unqualified, as noted in Specification 13-EN-700, "Installation Specification for the installation of NQR Maintenance and Monitoring Equipment for the Palo Verde Nuclear Generating Station Units 1, 2, and 3 - Quality Class Q, QAG, and NQR," Revision 1. The licensee initiated a WO to remove the unqualified tie-wraps. The licensee generated CRDR 2656591 to address this issue. The licensee performed additional inspections of the Units 1 and 3 ECCS sumps. The licensee identified and removed unqualified tie-wraps in Unit 1 ECCS sump Trains A and B and in Unit 3 ECCS sump Train B.

Analysis. This finding is greater than minor, since it affected the mitigating system cornerstone objective of equipment reliability because the debris could have affected containment spray pump flow by clogging spray nozzles. The finding is of very low safety significance because the amount of debris would have only degraded containment spray pump flow during a potential large break LOCA, but the containment spray system safety function would have been fulfilled based on the small amount of debris. This finding was screened as Green using Significance Determination Process Phase 1.

Enforcement. Technical Specification Surveillance Requirement 3.5.3.8 requires, in part, to verify by visual inspection that each ECCS train containment sump suction inlet is not restricted by debris. Step 8.4 of Procedure 31ST-9SI01 requires, in part, that personnel inspect the sumps and remove any foreign debris from the sump suctions. Contrary to these requirements, personnel did not remove unqualified tie-wraps from the Unit 1 ECCS sumps on October 25, 2002, the Unit 2 ECCS sumps on April 12, 2002, and Unit 3 ECCS sump Train B on April 24, 2003. Because this failure to remove debris was of very low safety significance and has been entered into the corrective action program via CRDR 2655298, this violation is being treated as an noncited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000528, 529, 530/2003005-01).

2. Auxiliary Feedwater Discharge Check Valve Test Failure

On December 8, 2003, operations personnel performed Procedure 73ST-9XI38, "AFA-P01 Discharge Checkvalve AFA-V015 - Inservice Test," Revision 5, in preparations for Mode 3 entry. Procedure 73ST-9XI38 is required following each time the check valve has been stroked open to maintain operability of auxiliary feedwater Pump B. The acceptance criteria of Procedure 73ST-9XI38, used to determine if check Valve AFA-V015 was fully seated, was not met to establish operability of auxiliary feedwater Pump B. Engineering determined that the surveillance test was invalid since plant conditions had not been established for performance of the test, since the discharge check valve had not been opened with pump flow. Further, engineering determined that no flow had passed through the check valve since the last successful test performance; therefore, the previously performed surveillance was valid to establish operability of auxiliary feedwater Pump B. Thus, engineering concluded that auxiliary feedwater Pump B was operable, Mode 3 could be entered, and that the check valve test did not need to be performed until after auxiliary feedwater Pump A was operated in Mode 3.

The inspectors continued their assessment of this issue at the end of this inspection period. Therefore, Unresolved Item (URI) 05000529/2003005-02, "Auxiliary Feedwater Discharge Checkvalve Test Failure," has been initiated pending completion of this review.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed control room operators' performance during simulator training on November 13, 2003, to evaluate emergency response by focusing on the risk-significant activity of classification. The inspectors also assessed operator recognition of abnormal plant conditions and implementation of the emergency plan.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

To review and assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, and high radiation areas, the inspector interviewed supervisors, radiation workers, and radiation protection personnel that had the potential to be involved in high dose rate and high exposure jobs during routine and Refueling Outage 2R11 operations. The inspector discussed changes to the access control program with the radiation protection superintendent. The inspector also conducted plant walkdowns within the radiologically controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area postings, radiation exposure permits, radiological surveys, and other controls for airborne radioactivity areas, radiation areas, and high radiation areas
- High radiation area key control
- Internal dose assessment for exposures exceeding 50 mrem committed effective dose equivalent (none observed during this inspection period)
- Setting, use, and response of electronic personal dosimeter alarms
- Conduct of work by radiation protection technicians and radiation workers in areas with the potential for high radiation dose and the associated radiation exposure permits and radiological surveys, and controls for the work associated with Refueling Outage 2R11 activities (Radiation Exposure Permits 2-325D, "Inspection/Cleaning of Reactor Vessel In core Detector Penetration"; 2-047A, "Reactor Vessel Closure Head Insulation Modification and Inspection"; 2-400A, "Pressurizer Heater Sleeve Cut Out and Replacement," and 2-501F, "RP Tours, Inspections, and Routine Surveys"

- Dosimetry placement when work involved a significant dose gradient (Radiation Exposure Permits 2-325D, "Inspection/Cleaning of Reactor Vessel In core Detector Penetration"; 2-047A, "Reactor Vessel Closure Head Insulation Modification and Inspection"; and 2-400A, "Pressurizer Heater Sleeve Cut Out and Replacement"

- Controls involved with the storage of highly radioactive items in the spent fuel and refuel pools

- Audits, licensee event reports (LERs), special reports, and self-assessments involving high radiation area controls and staff performance (no LERs or special reports were reviewed during this inspection period)

- Summary of corrective action documents written since the last inspection and selected documents related to high radiation area incidents, radiation protection technician and radiation worker errors, and repetitive and significant individual deficiencies

Performance indicator (PI) reviews are documented in Section 4OA1 of this report.

In addition, the inspector reviewed the licensee's respiratory protection program for compliance to 10 CFR 20.1703(f) requirements.

b. Findings

No findings of significance were identified.

2OS2 As Low as is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

To assess the licensee's program to maintain occupational exposures ALARA, the inspector reviewed work packages conducted during Refueling Outages 2R10, 1R10, 3R10, and 2R11. In addition, the inspector attended the prejob ALARA brief and observed radiological work associated with the movement of three low level waste drums that created locked high radiation area conditions.

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. In addition, the following items were reviewed and compared with procedural and regulatory requirements:

- Current 3-year rolling average collective exposure

- Thirteen ALARA prejob, in-progress, and postjob reviews and associated radiation exposure permit packages from the past four refueling outages, which resulted in some of the highest personnel collective exposures

- Site-specific trends in collective exposures, historical data, and source term measurements
- Site-specific ALARA program procedures
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Work activity regarding intended dose against actual dose received and the reasons for any inconsistencies
- Assumptions and basis for annual collective exposure estimates, the methodology for estimating work activity exposures, and intended dose outcomes
- 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
- Historical trends and current status of tracked plant source terms and contingency plans due to changes in fuel performance or primary plant chemistry
- Radiation worker performance during work activities in radiation, high radiation, or airborne radioactivity areas
- Two declared pregnant workers' declarations during the assessment period and monitoring controls and exposure result
- Self-assessments and audits related to the ALARA program since the last inspection
- Resolution through the corrective action process of problems identified through postjob reviews and postoutage report critiques
- The effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies
- Summary of corrective action documents written since the last inspection and selected documents relating to exposure tracking, higher than planned exposure levels, radiation worker practices, and repetitive and significant individual deficiencies against the corrective action program

The inspector completed 15 required and one additional sample requirements.

b. Findings

No findings of significance were identified.

4OA1 PI Verification (71151)

a. Inspection Scope

The inspectors verified the accuracy of the PI data reported and used the PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-2, "Regulatory Assessment Indicator Guideline," Revision 2, to verify the basis in reporting for each data element.

Barrier Integrity Cornerstone

- Reactor Coolant System (RCS) Specific Activity (Units 1, 2, and 3)

The inspectors reviewed a random sample of the RCS activity data logs from December 2002 through November 2003 to verify the accuracy and completeness of data associated with the RCS specific activity reported for all three units.

- RCS Identified Leak Rate (Units 1, 2, and 3)

The inspectors reviewed the licensee's RCS leakage database from December 2002 through November 2003 to verify the accuracy and completeness of the data associated with the RCS leakage reported for all three units.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness PI

Licensee records reviewed included corrective action documentation that identified occurrences of locked high radiation areas (as defined in Technical Specification 5.7.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled.

Public Radiation Safety Cornerstone

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

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

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

Annual Sample Review

The following three annual samples were reviewed in accordance with Inspection Procedure 71152:

1. Review of Control Rod Bottom Light Indication Problems

a. Inspection Scope

The inspectors reviewed the following CRDRs 2627031, 2557409, 2489698, and 2430998 related to intermittent malfunction of the rod bottom indication lights. The inspectors also interviewed the cognizant system engineer regarding troubleshooting activities. The documents were reviewed to determine if adequate corrective actions were performed.

b. Findings

The inspector noted that the interviews and CRDRs showed that the problems in the rod bottom indicator lights appeared randomly on reactor trips, often resolved before troubleshooting could determine a cause, and did not violate any requirement. Conservative actions were taken by operators in response, such as boration initiation or alternate rod bottom position determination. The cause was finally determined to be apparent intermittent high resistance on contacts of relays in the rod bottom light indication system and these relays are being replaced by new sealed models. There were no findings of significance identified.

2. Improper Valve Line-ups

a. Inspection Scope

The inspectors selected two CRDRs for detailed review (CRDRs 2597828 and 2648597). The two issues described in the CRDRs pertain to improper valve line-ups on plant systems. The inspectors performed this evaluation to identify similarities, if any, between the two causes of the deficient conditions.

On April 15, 2003, inspectors identified three valves out of position on Unit 3 EDG Train B. The cause of the misaligned valves was inadequate configuration change management during implementation of a plant modification. Specifically, the operating procedure was modified to reflect the correct valve configuration; however, the technical document used to restore the system to an operable line-up was not modified. Corrective actions included improvements to the design change management process.

On October 29, 2003, a loss of instrument air occurred shortly after returning the system to service following planned outage work. The cause of the instrument air header pressure reduction was a mispositioned valve. This configuration control error occurred due to inadequacies in both the technical document and operating procedure with regard to valve position upon system restoration or establishing a normal lineup. The procedure errors were a result of an inadequate procedure change process.

b. Findings

The inspectors verified that the two causes for the improper valve line-ups were unrelated and that corrective actions from the April 15 issue should not have prevented the event on October 29. The inspectors further verified that the cause evaluations and associated corrective actions were appropriate and also timely, relative to the identified problems; therefore, no violation of regulatory requirements or findings were identified.

3. Main Steam Line - Bolts Missing

a. Inspection Scope

The inspectors reviewed the licensee’s corrective actions associated with identifying anchor bolt nuts missing from a knee brace anchor that supported a Unit 2 main steam line pipe whip restraint. The inspectors reviewed CRDR 2643347 to verify the licensee identified the full extent of the issue, performed appropriate evaluations, and specified suitable corrective actions.

b. Findings

Introduction. An apparent self-revealing violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for which the significance has yet to be determined. The performance deficiency resulted in the failure to install anchor bolt nuts supporting a main steam line pipe whip restraint for Unit 2 Steam Generator 2.

Description. On October 13, 2003, during removal of Unit 2 Steam Generator 2, personnel discovered that a main steam line pipe whip restraint knee brace was not properly secured to a structural building wall inside containment. Specifically, it was noted that four 1/2-inch diameter nuts were found missing from the embedded anchor bolts as shown in Drawings 13-C-ZCS-541, Section B, and 13-C-ZCS-542, Section D. These anchor bolts and nuts provided structural support for Pipe Whip Restraint Hanger 02-SG-042-H-890. The licensee placed this degraded condition into the corrective action process as CRDR 2643347.

The inspectors reviewed CRDR 2643347 and discussed the potential consequences of the degraded condition with engineering personnel. The inspectors noted that the licensee had initially determined that the condition could potentially result in failure of the steam line containment penetration, the containment liner, or the containment spray headers due to pipe whip following a postulated main steam line break accident. Due to the potential significance of this condition, the licensee assigned an outside contractor to perform a detailed analysis of the condition. Although at the time of the inspection the analysis was not complete or provided for NRC's review, the licensee stated that preliminary results obtained by the contractor analysis indicate that the as-found degraded condition would not result in the main steam line affecting operability of the containment spray system, containment liner, or main steam line containment penetration. The licensee stated that the analysis is scheduled to be completed and provided for NRC's review around January 31, 2003.

The inspectors noted that the licensee had performed a review of maintenance activities and modifications performed that could have potentially resulted in this degraded condition. The licensee noted that no activities had been performed that would have affected the restraint knee. An inspection of the structural steel and surrounding concrete was performed and it was determined that there was no indication that the nuts were ever installed and quality control checked since initial placement of the restraint knee.

<u>Analysis</u>. The inspectors determined that this performance deficiency was more than minor because it affected the barrier integrity cornerstone to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors reviewed the finding using Inspection Manual chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The deficiency was determined to result in a potential open pathway affecting the physical integrity of reactor containment requiring use of Inspection Manual chapter 0609, Appendix H, "Containment Integrity." The inspectors concluded the containment integrity function would only be affected during a postulated main steam line break scenario and that the exposure time for this degraded condition, coupled with the probability of core damage, would require further review in order to determine the risk significance of the issue. The inspectors determined that review of the analysis the licensee was having performed by a contractor would also provide additional information relative to the significance of the issue.

<u>Enforcement</u>. 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawing," requires, in part, that activities affecting quality shall be prescribed by documented drawings and shall be accomplished in accordance with these drawings. The inspectors determined that the failure to secure Unit 2 main steam line knee brace for Pipe Whip Restraint Hanger 02-SG-042-H-890 in accordance with Drawings 13-C-ZCS-541, Section B, and 13-C-ZCS-542, Section D, was an apparent violation of 10 CFR Part 50, Criterion V. The licensee documented this deficiency in their corrective action process as CRDR 2643347. This finding does not represent an immediate safety concern because new anchor bolt nuts were installed during reassembly of the system as part of the Unit 2 steam generator replacement program. Pending determination of its risk significance, the apparent violation is identified as URI 05000529/2003005-03, "Missing Bolts on Support for Main Steam Line Whip Restraint."

Cross-Reference to PI&R Findings Documented Elsewhere

Section 2OS1 evaluated the effectiveness of the licensee's problem identification and resolution processes related to high radiation area incidents and radiation protection technician and radiation worker errors. No findings of significance were identified.

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. No findings of significance were identified.

Quarterly Review

The inspectors reviewed a selection of condition reports 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 condition reports 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.

4OA3 Event Followup

1. Heavy Load Drop in Containment

a. Inspection Scope

The inspectors observed licensee response to the drop of a 7000-pound steam generator snubber lever plate in containment during a rigging operation. This event occurred on October 3, 2003, during the Unit 2 steam generator replacement outage. 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 inspectors assessed the significance of the heavy load drop and any actual and potential plant impacts that resulted from the event.

b. Findings

The safety significance of this event did not meet the criteria for a special inspection. The inspectors evaluated this event in detail and the results are documented in NRC Inspection Report 05000529/2003009, Sections 4OA2.4 and 4OA5.1.

2. Fuel Sipping Canister Drop

a. Inspection Scope

The inspectors observed licensee response to the drop of a fuel sipping canister that fell approximately 30 feet onto the sipping stand when its rigging came loose on October 23, 2003. The fuel sipping canister was empty and was being removed from the Unit 2 cask loading pit to swap it with another functioning canister. There was no fuel sipping in progress as activities had been previously suspended due to a malfunctioning canister lid actuating cylinder. Further, all spent fuel was in designated storage racks and isolated from the cask loading pit by the installed gate. As part of the followup to this event, the inspectors observed initial operator response and immediate corrective actions that were implemented and reviewed CRDR 2645823, which included personnel statements.

b. Findings

No findings of significance were identified.

3. EDG Train A Engineered Safety Feature Actuation

a. Inspection Scope

The inspectors reviewed licensee response to a loss of power (LOP) to 4.16 kV safety-related Bus 2EPBAS03 on November 21, 2003. EDG Train A was supplying the safety-related bus as the sole source of power when the EDG output breaker unexpectedly tripped open during testing per Procedure 40T-9GT01, "GTG Isochronous Test," Revision 0. The electrical bus lost all power, resulting in a valid LOP signal based on the undervoltage condition. The LOP emergency autostart signal was received by EDG 2A, and its output breaker reclosed back onto the safety-related bus within a few seconds as per design. The following information was reviewed and used as criteria for evaluating response to this event:

- Event Notification 40349
- CRDRs 2654236 and 2655188
- Sequence of events
- Troubleshooting activities and results per Work Mechanism 2654431, Revision 1
- Procedure 40OP-9DG01, "Emergency Diesel Generator A," Revision 30
- Procedure 40TI-9GT01, "GTG Isochronous Test," Revision 0

b. Findings

No findings of significance were identified.

4. (Closed) LER 05000528/2003001-00, Pressurizer Safety Valve As-Found Lift Pressure Outside of Technical Specification Limits

On March 5, 2003, set pressure verification testing was completed on the four pressurizer safety valves that had been removed during the Unit 1 tenth refueling outage. The testing revealed that the as-found set pressure of one of the four valves was greater than the maximum allowable set pressure listed in the Technical Specifications. The function of the pressurizer safety valves is to limit reactor coolant pressure to less than or equal to the Technical Specification safety limit pressure of 2750 psia for moderate to low frequency events and to less than 3000 psia for very low frequency events. The inspectors noted that the Technical Specification required as-found set pressure for the valves is a nominal 2475 psia with a tolerance of plus 3 percent, minus 1 percent. The as-found set pressure was 2550 psia or 3.7 percent of design lift pressure. The licensee evaluated the impact of the pressurizer safety valve out-of-tolerance set pressure and determined that the results were bounded by the peak reactor coolant pressure results of the loss of condenser vacuum analysis. The licensee concluded that the safety function of the pressurizer safety valves would have been met.

The inspectors reviewed CRDR 2589790 and it's significant root cause investigation. The inspectors noted that the licensee found that the valve spring was not square due to age. Once the licensee replaced the old spring with a new one, they found that the valve had a consistent set pressure with the new spring. The licensee's corrective action was to measure additional parameters for each valve spring after being in service for a cycle. This finding constituted a violation of minor significance that is not subject to enforcement action in accordance with Section VI of the NRC's Enforcement Policy. The licensee documented the problem in CRDR 2589790. This LER is closed.

4OA5 Other Activities

1. Reactor Pressure Vessel (RPV) Head and Vessel Head Penetration Nozzles (TI 2515/150)

Susceptibility Ranking Calculation

a. Inspection Scope

On October 6-10, 2003, the inspectors performed NRC Inspection Manual Temporary Instruction 2515/150 for Unit 2 during Cycle 11 Refueling Outage 2R11. They reviewed the licensee's inspection plan in response to NRC Order EA-03-009 (Order) which established interim inspection requirements for RPV heads.

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

The inspectors noted that Unit 2 was in the moderately susceptible category and that the plant had no previous inspection findings requiring classification as high susceptibility. Required inspections for refueling outage were bare metal visual examination of 100 percent of the RPV head surface (Order Section IV.C.(2)(a)), ultrasonic testing of each RPV head penetration nozzle from 2 inches above the

J-groove weld to the bottom of the nozzle (Order Section IV.C.(2)(b)(i)), or eddy current testing of the wetted surface of each J-groove weld and RPV head penetration nozzle base material to at least 2 inches above the J-groove weld (Order Section IV.C.(2)(b)(ii)). Because of hardships, the licensee had, with the ability to perform inspections in strict compliance with the Order, two relaxation requests submitted to the NRC and approved based on the demonstration of good cause for the proposed relaxations. The first proposed alternative examination was to perform a bare metal visual examination of the one RPV head vent line nozzle in accordance with Order Section IV.C.(2)(a), since internal volumetric or surface examination would be difficult and would require the removal of the welded orifice and testing of the remaining control element drive mechanism nozzles per Order Section IV.C.(2)(b). The second proposed alternative examination was to perform ultrasonic testing of each nozzle from 2 inches above the J-groove weld to approximately 0.6 inches above the top of the nozzle's chamfer face control element drive mechanism since ultrasonic scans in the area below 0.6 inches to the bottom of the nozzle do not yield useful data because of the geometry of the nozzle and funnel.

b. Findings

No findings of significance were identified.

Volumetric and Surface Examinations

a. Inspection Scope

The inspectors verified that the licensee's volumetric inspection plan and critical performance objectives were incorporated into site procedures. They also interviewed plant inspection personnel, and contractors performing the inspections, to determine their understanding of inspection standards and acceptance criteria required during data gathering and analysis. The inspectors reviewed the Westinghouse Field Service Procedures which governed the instrument calibration, data gathering, and data analysis requirements for ultrasonic and eddy current testing. Nuclear Reactor Regulation personnel, in conjunction with the inspectors, reviewed the qualification of these methods and their ability to determine flaws in J-groove welds and base metals associated with primary water stress corrosion cracking. The inspectors reviewed licensee and contractor qualifications and certification records which were obtained through a combination of written and practical examinations. The inspectors conducted interviews with plant engineers and Westinghouse contractors to determine their training, background, the basis used for certifications, and expertise in conducting and analyzing these examinations. The inspectors also observed equipment operation during data gathering and data analysis for a sample of head penetration nozzles to assess procedural adherence.

b. Findings

No findings of significance were identified.

Bare Metal Visual Examinations

a. Inspection Scope

The inspectors observed the video acquired during visual inspection of the RPV head vent line nozzle and noted that the camera and remote monitoring equipment used during the examination process provided adequate visual clarity. The inspectors reviewed certification records and discussed the qualifications and experience of the examiners. The inspectors verified that a clear 360^{o} observation of the nozzle was completed and that no evidence of cracking or boric acid crystals were present.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On October 8-10, 2003, the inspectors reviewed the licensee’s response to NRC Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity." The response described the licensee’s commitment to perform a bare metal visual inspection of all 62 nozzle penetrations in the lower reactor head of all three units. The inspector reviewed the licensee's procedures for the inspection of the Unit 2 lower head penetrations. The inspector also reviewed the qualification and certifications for the personnel performing the inspections.

The inspectors reviewed a video tape of the nozzle inspections that covered full 360^{o} coverage of all 62 nozzle penetrations. The licensee identified 26 nozzles that had an excessive amount of a "spraylat" which prevented adequate assessment of the penetration condition. The licensee determined that the spraylat was a protective latex coating that was installed for transport of the vessel during construction. The licensee stated that they would remove the spraylat and re-examine the identified nozzles at a later date in the refueling outage.

The inspectors observed a portion of the spraylat removal process. The spraylat was removed by a dry-ice impingement process. The inspectors reviewed the final inspection of the cleaned nozzles and determined that there was no boric acid. The licensee analyzed a sample of the latex removed by the cleaning process and determined that it did not contain boric acid.

b. Findings

No findings of significance were identified.

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

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

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

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

The inspector also assessed any sump-related modifications.

Interim Compensatory Measures

a. Inspection Scope

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

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

b. Findings

The licensee informed operators of the actions for potential blockage but decided to defer any procedure changes until the Combustion Engineering Owners Group evaluates the bulletin. The licensee is also considering a modification to doors in the bioshield, has initiated a design change request, but has not decided on any final design changes. The licensee did look for debris during walk down of the containment during the previous Unit 2 outages. The location of the debris and assessment is noted in Calculation 13-MC-SI-309, "Containment Sump Blockage," Revision 3. The inspectors toured containment and did not see any insulation that was not previously identified.

No findings of significance were identified.

Debris Sources in Containment

a. Inspection Scope

The potential debris sources in containment are described in UFSAR, Section 6.2.2, and in Calculation 13-MC-SI-309. The inspectors also toured Unit 2 containment for this identified potential debris source to verify that there were no additional debris sources.

b. Findings

No findings of significance were identified.

Containment Sump Inspection and Design

The inspection purpose is to assist the staff in determining whether additional measures are warranted to ensure that PWR licensees are performing containment walkdowns for debris sources in a timely manner. NEI has guidance entitled NEI 02-01, Revision 1, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," dated September 2002.

a. Inspection Scope

The inspectors reviewed the design of the containment sumps which are designed to be reservoirs of water to the ECCS following a LOCA. The design requirements for the sumps are to filter the RCS water to preclude particles greater than 3/16-inch diameter from entering the ECCS sump.

b. Findings

Introduction. The inspectors identified a noncited violation related to 10 CFR Part 50, Criterion III, "Design Control." This violation is related to the identification of an unscreened hole on the ECCS sump covers for Trains A and B. These 1-inch holes were greater than the 1/8-inch gap allowed by the ECCS sump design.

Description. On December 4, 2003, during closeout inspections for the containment sumps, the inspectors identified an approximately 1-inch hole on each ECCS sump covers for Trains A and B. During initial construction of the ECCS sumps, temperature instruments were placed in these in accordance with RG 1.97, "Post-Accident Instrumentation," per Design Change (DCP) 10J SI 156. Holes were drilled to allow a conduit through the ECCS sump cover. Due to interference with flanges below the cover where the holes were initially drilled, an additional hole was drilled into each ECCS sump cover to relocate the instruments and allow for this new conduit. The first drilled holes were not filled and left unscreened. Notes on Drawing 13-C-ZCS-669 note that holes greater that 3/8-inch are to be screened and gaps in the sump cover plates are only allowed to be 1/8-inch wide.

The licensee initiated CRDR 2656229 to address this issue. Unit 3 was also subject to modification DCP 10J SI 156; therefore, the licensee performed an inspection of the Unit 3 ECCS sump covers for Trains A and B. No holes on the Unit 3 ECCS sump covers were identified. The licensee's immediate corrective action was to install bolts with washers on each Unit 2 ECCS sump covers for Trains A and B.

Analysis.

This finding is greater than minor because it affected the mitigating system cornerstone objective of equipment reliability by not assuring that the ECCS sump structure would filter out all debris greater than 3/16-inch diameter. Thus, debris entering the ECCS sump could have potentially affected the safety injection system and the containment spray system following a postulated LOCA. The finding is of very low safety significance because the location of these holes were not in the design flowpath for water into the ECCS sump, which would have limited the amount of debris introduced into the system. There is an operational platform above the sumps, which would also limit debris falling onto the sump cover. This finding was screened as Green using Significance Determination Process Phase 1.

Enforcement. 10 CFR Part 50, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis . . . for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. Drawing 13-C-ZCS-669, "Containment Internals - Emergency Recirculation Sump Screen Plans, Sections and Details," Revision 6, notes that holes greater that 3/8-inch are to be screened, and gaps in the sump cover plates are only allowed to be 1/8-inch wide. Contrary to the above, DCP 10J SI 156 completed on May 22, 1985, incorporated a 1-inch unscreened hole on each of ECCS sump covers for Trains A and B. Because this finding was determined to be of very low safety significance and has been entered into the corrective action program via CRDR 2656229. This violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000529/2003004-04).

4. Institute of Nuclear Power Operations Report Review

a. Inspection Scope

The inspectors reviewed the Institute of Nuclear Power Operations assessment dated August 2003.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

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.

The resident inspectors presented the inspection results to Mr. G. Overbeck, Senior Vice President, Nuclear, and other members of licensee management during an exit interview conducted on January 7, 2003.

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

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Bauer, Department Lead, Regulatory Affairs
J. Bayless, Inservice Inspection Engineer
P. Borchert, Director, Work Management
R. Buzard, Regulatory Affairs
D. Carnes, Director, Regulatory Affairs and Nuclear Assurance
K. Coon, Technical Assistant, Radiation Protection
M. Fladager, Department Leader, Radiation Protection
J. Gaffney, Director, Radiation Protection
F. Gowers, Site Representative, El Paso Electric
T. Gray, Department Leader, Radiation Protection
D. Hanson, Inservice Inspection Engineer
D. Hautala, Licensing Engineer
R. Henry, Site Representative, Salt River Project
R. Indap, Inservice Inspection Engineer
J. Levine, Executive Vice President, Generation
D. Marks, Section Leader, Regulatory Affairs
D. Mauldin, Vice President, Engineering and Support
G. Michael, Regulatory Affairs
M. Milton, Section Lead, Inservice Inspection Engineer
G. Overbeck, Senior Vice-President
S. Peace, Consultant, Communications
M. Powell, Department Lead, Maintenance Engineering
T. Radtke, Director, Operations
D. Smith, Plant Manager, Production
M. Sontag, Department Lead, Nuclear Assurance
M. Winsor, Director of Engineering

Others

F. Gowers, Site Representative, El Paso Electric
R. Henry, Site Representative, Public Service of New Mexico

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000528, 529, 530/2003005-01	NCV	Tie Wraps in the ECCS Sumps (Section 1R22.1)

05000529/2003005-02	URI	Auxiliary Feedwater Discharge Checkvalve Test Failure (Section 1R22.2)
05000529/2003005-03	URI	Missing Bolts On Support for Main Steam Line Whip Restraint (Section 4OA2.3)
05000529/2003005-04	NCV	ECCS Sump Covers Not Maintained According to Design Drawings (Section 4OA5.3)

Closed

05000528, 529, 530/2003005-01	NCV	Tie Wraps in the ECCS Sumps (Section 1R22.1)
05000529/2003005-04	NCV	ECCS Sump Covers Not Maintained According to Design Drawings (Section 4OA5.3)
05000528/2003001-00	LER	Pressurizer Safety Valve As-Found Lift Pressure Outside of Technical Specification Limits (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

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

Section 1R01: Adverse Weather

Procedures

40OP-9ZZ17, "Cold Weather Protection," Revision 21

CRDR

2637745

Section 1R04: Equipment Alignment

Procedures

40ST-9SI13, "Low Pressure Safety Injection System Alignment Verification," Revision 3

Drawings

02-M-SIP-001, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 23
02-M-SIP-002, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 21

01-M-SIP-001, "P&I Diagram, Safety Injection and Shutdown Cooling System," Revision 25
03-M-SGP-001, Sheet 2, "P&I Diagram, Main Steam System," Revision 42

Miscellaneous

Tag Assignment Sheet 98755

Section 1R06: Internal Flood Protection

Procedures

PVNGS Design Basis Manual C2, "Hazards Topical," Revision 6

Calculations

13-MC-ZA-808, "MSSS Flooding at Elevation 81 Feet," Revision 2
13-MC-DG-204, "Diesel Generator Building Flooding Analysis," Revision 1

Section 1R08: Inservice Inspection Activities

Procedures

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

Miscellaneous Reports

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

Design Modification WO

DMWO 2513813, "Modification to the Pressurizer Heater Sleeves," Revision 0

10 CFR 50.59 Screening

E-03-0005, "Pressurizer Heater Sleeves," Revision 0

Section 1R11: Licensed Operator Requalification

Miscellaneous Documents Reviewed

Table to Gregory Werner from Joe Allison on October 22, 2003, providing the 2003 annual operating examination results.

Section 1R12: Maintenance Implementation

CRDR

2644780

WO

2326439

Miscellaneous

Technical Instruction Manual, CENTM-16, "Operation - Maintenance Instructions for Ex-Core Neutron Flux Monitoring System," Revision 1

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

WO

P-MS1-001
2585437

Procedures

S.D.-OP-1994-6675, "PWR/BWR Canister Sipping System Operating Procedure," Revision 8
40OP-9PC07, "Miscellaneous Fuel Pool Operations," Revision 32
33MT-9ZZ02, "Freeze Sealing," Revision 5

Miscellaneous

"N.M. Evaluation of Fuel Sipping Activities for U2R11," Revision 1

Section 1R15: Operability Evaluations

Miscellaneous

10 CFR 50.59 Screening S-03-0297, "DMWO 2645454," Revision 0

1R17: Permanent Plant Modifications

Design Modification WOs

2579336
241003

Engineering Document Changes

2001-00476
2003-00196

Section 1R19: Postmaintenance Testing

CRDR

2636177

Section 1R20: Refueling and Outage Activities

Procedures

Procedure 72IC-9RX03, "Core Reloading," Revision 22
Procedure 40ST-9ZZ09, "Containment Cleanliness Inspection," Revision 7
Procedure 72PY-9RX04, "Low Power Physics Testing using RMAS," Revision 5

Permits

91200, "1281 2MDGAH01 Master U2R11"
93916, "1652 Install PW System T-Mod 02"
91346, "Inspection of the Train A Containment Sump"
91347, "Inspection of the Train B Containment Sump"
91356, "1691 Core Reload Permit"
94106, "Pressurizer Heater PM's"
99584, "2PSIBV532 Intrusive Rework"

CRDR

2657004

Section 2OS1: Access Control to Radiologically Significant Areas

Quality Assurance Audits and Surveillances

Nuclear Assurance Audit 2002-008, "Radiation Safety"
Self Assessment, August 26, 2003, "LHRA Controls during U3R10"
Self Assessment, June 17-27, 2003, "Intakes: Review and Documentation Trail"
Self Assessment, July 15-30, 2003, "Radiation Worker Error Tracking and Trending"
Nuclear Assurance Evaluation Report ER 02-0101
Nuclear Assurance Evaluation Report ER 02-0102

CRDRs

2554246, 2559664, 2570518, 2596452, 2596757, 2605783, and 2626224

Procedures

01DP-0IS08, "PVNGS Respiratory Protection Equipment Usage," Revision 9

75RP-9OP01, "Radiological Controls for Diving Operations," Revision 6

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

75DP-9RP01, "Radiation Exposure and Access Control," Revision 5

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

75RP-9RP07, "Radiological Surveys," Revision 9

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

75RP-0LC02, "Performance Indicator Public Radiation Safety Cornerstone," Revision 0

75RP-0LC01, "Performance Indicator Instruction Guideline Occupational Radiation Safety Cornerstone," Revision 0

Section 2OS2: ALARA Planing and Controls

Quality Assurance Audits and Surveillances

Nuclear Assurance Evaluation Report ER 02-0292

Nuclear Assurance Evaluation Report ER 03-0146

Nuclear Assurance Evaluation Report ER 02-0005

Nuclear Assurance Evaluation Report ER 02-0101

Self-Assessment, February 28, 2003, Incorporation of ALARA Principles into Plant Modification Designs

Reports

2001 Annual Radiation Protection Program Summary Report
2002 Annual ALARA/Management Evaluation Report
Post Refueling Outage ALARA Report 3R10
Post Refueling Outage ALARA Report 2R10
Post Refueling Outage ALARA Report 1R10

Procedures

75DP-0RP06, "ALARA Committee," Revision 3
75DP-0RP03, "ALARA Program Overview," Revision 2
75DP-9RP01, "Radiation Exposure and Access Control," Revision 6, Section 3.10.4
75RP-9RP12, "ALARA Reports," Revision 1
75RP-9RP25, "Temporary Shielding," Revision 4
RP 9-76, "Reactor Coolant Source Term Trending"
Night Order 01-010, "Hot Spot Tracking, Trending, and Flushing," December 6, 2001

Radiation Exposure Permits

Unit 2R10

2-3502, "Valve, Flange and Pump Maintenance and Inspection"
2-3304, "Installation of Pressurizer Spray Line Shielding Frame"
2-3508, "In Service Inspection and Associated Work"

2-3412, "Pressurizer Heater Removal and Replacement"
2-3305, "Steam Generator Replacement Preparations"

Unit 1R10

1-3502, "Valve, Flange and Pump Maintenance and Inspection"
1-3002, "Reactor De-Stack and Re-Stack"

Unit 3R10

3-3306, "Primary Side Steam Generator Maintenance"
3-1325, "Pressurizer Heater Replacement and Nozzle Repair"
3-3502, "Valve, Flange and Pump Maintenance and Inspection"

Unit 2R11

2-3047, "Reactor Vessel Head Insulation Modification and Inspection"
2-3325, "Inspection/Clean Reactor Vessel Incore Detector Penetrations"
2-3400, "Pressurizer 100 Percent Heater Sleeve Replacement"

CRDRs

2603646, 2600110, 2599068, 2505215, 2567062, 2575217, 2588954, 2591773, 2655725, 2656105, 2648595, and 2651332

Section 4OA3: Event Followup

Significant CRDR 2589790

Section 4OA5: Other Activities

Manuals

Nuclear Administrative and Technical Manual 73TI-9ZZ78, "Visual Examination for Leakage"

Nuclear Administrative and Technical Manual 73DP-9ZC01, "Boric Acid Corrosion Control Program"

Drawings

13-C-ZCS-301, "Containment Internals - Partial Concrete Plan at El. 80’-0" Areas CAC and CAD," Revision 17

CRDR

2656229

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
CPC/CEAC	core protection calculators and control element assembly calculator
CRDR	condition report/disposition requests
ECCS	emergency core cooling system
EDG	emergency diesel generator
LER	licensee event report
LOCA	loss of coolant accident
LOP	loss of power
NCV	noncited violation
NEI	Nuclear Energy Institute
PI	performance indicator
PWR	pressurized water reactor
RCS	reactor coolant system
RPV	reactor pressure vessel
TBD	to be determined
UFSAR	updated final safety analysis report
URI	unresolved item
WO	work orders