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

February 4, 2008

Randall K. Edington
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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000528/2007005, 05000529/2007005, AND 05000530/2007005

Dear Mr. Edington:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palo Verde Nuclear Generating Station, Units 1, 2, and 3, facility. The enclosed integrated report documents the inspection findings, which were discussed on January 9, 2008, 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 four NRC identified findings and two self-revealing findings. These findings were evaluated under the risk significance determination process as having very low safety significance (Green). Because of the very low safety significance of these violations and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations consistent with Section VI.A of the NRC Enforcement Policy. Five licensee-identified violations, which were determined to be of very low safety significance, are listed in Section 4OA7 of this report. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, facility.

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

Sincerely,

/RA/

Troy W. Pruett, Chief
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 05000528/2007005, 05000529/2007005, and 05000530/2007005
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SUNSI Review Completed:__TWP__ADAMS: ☒ Yes ☐ No Initials: _TWP_____
☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive
R:_REACTORS_PV\2007\PV2007-005RP-GGW.wpd ADAMS ML080350669

RIV:RI:DRP/D	RI:DRP/D	RI:DRP/D	SRI:DRP/D	SPE:DRP/D
JFMelfi	MCatts	JBashore	GGWarnick	GEWerner
/RA/	/RA/	/RA/	/RA/	GEW
1/31/08	1/31/08	1/31/08	1/31/08	1/30/08

C:DRS/EB2	C:DRS/PSB	C:DRS/OB	C:DRS/EB1	C:DRP/D
LJSmith	MPShannon	RLantz	RLBywater	TWPruett
CFOKeefe for	KEBrooks for	TOMcKernon for	RLB	TWP
1/30/08	1/30/08	1/30/08	1/30/08	2/1/08

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-528, 50-529, 50-530
Licenses:	NPF-41, NPF-51, NPF-74
Report:	05000528/2007005, 05000529/2007005, 05000530/2007005
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Road Tonopah, Arizona
Dates:	October 1 through December 31, 2007
Inspectors:	J. Bashore, Resident Inspector M. Bloodgood, Reactor Engineer L. Carson, Senior Health Physics Inspector M. Catts, Resident Inspector G. George, Reactor Inspector, Engineering Branch 1 C. Graves, Health Physics Inspector B. Henderson, Reactor Inspector, Engineering Branch 1 T. McKernon, Senior Operations Inspector, Operations Branch S. Makor, Reactor Inspector, Engineering Branch 1 J. Melfi, Resident Inspector L. Ricketson, P.E., Senior Health Physics Inspector G. Warnick, Senior Resident Inspector
Approved By:	Troy W. Pruett, Chief, Project Branch D Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000528/2007005, 05000529/2007005, 05000530/2007005; 10/01/07 - 12/31/07; Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Integrated Resident and Regional Report; Maintenance Effectiveness, Operability Eval., Access Control to Radiological Areas, ALARA Planning, Identification and Resolution of Problems.

This report covered a 3-month period of inspection by resident inspectors and regional inspectors. The inspection identified six 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," was identified for the failure of engineering personnel to promptly correct a significant condition adverse to quality. Specifically, on September 17, 2007, the steam supply to auxiliary feedwater Pump A bypass Valve SGA-UV-138A failed to open as required during the performance of the quarterly surveillance test. The cause of the failure was determined to be foreign material on the valve's internal components. Corrective actions were implemented but the source of the debris was not definitively identified. Subsequently, on October 15, 2007, the valve failed to close. Further investigation indicated that the failure was caused by foreign material on the valve's internal components. This issue was entered into the corrective action program as Condition Report/Disposition Request 3078032.

  The finding is greater than minor because a failure to open is associated with the equipment performance attribute of the mitigating systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, a failure to close is associated with the structure, system, and component and barrier performance attribute of the barrier integrity cornerstone and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, this finding is determined to have very low safety significance because the finding did not result in a loss of safety function under the mitigating systems cornerstone and did not result in an actual open pathway in the physical integrity of the reactor containment under the containment barrier cornerstone. This finding has a crosscutting aspect in the area of human performance associated with work control because the facility did not dedicate the manpower and expertise necessary to coordinate work activities to incorporate actions to support long term equipment reliability and safety system availability [H.3(b)] (Section 1R12).

- Green. The inspectors identified two examples of a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the failure of operations personnel to follow procedures and adequately evaluate degraded and nonconforming conditions to support operability decision-making. On September 12 and October 29, 2007, operations personnel failed to adequately evaluate degraded and nonconforming conditions to support operability decision-making as described in Procedure 40DP-9OP26. Specifically, operations personnel failed to adequately evaluate the operability of the Unit 2 Train B emergency diesel generator after a lowering turbocharger lube oil pressure indication and the Unit 1 Train A auxiliary feedwater system during a body to bonnet steam leak on manual isolation Valve SGE-V886 (steam supply to auxiliary feedwater Pump A bypass Valve SGA-UV-138A). This issue was entered into the corrective action program as Condition Report/Disposition Request 3068929 and Palo Verde Action Request 3084439.

  The finding is greater than minor because the degraded turbocharger lube oil filter is associated with the equipment performance attribute of the mitigating systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, the steam leak on manual isolation Valve SGE-V886 is associated with the structure, system, and component and barrier performance attribute of the barrier integrity cornerstone and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding is determined to have very low safety significance because the finding did not result in a loss of safety function under the mitigating systems cornerstone and did not result in an actual open pathway in the physical integrity of the reactor containment under the containment barriers cornerstone. The example of this finding related to lowering turbocharger lube oil pressure has a crosscutting aspect in the area of human performance associated with decision-making because the licensee did not use conservative assumptions for operability decision-making when evaluating degraded and nonconforming conditions [H.1(b]). The example of this finding related to the body to bonnet steam leak has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not properly classify, and thoroughly evaluate the operability for a condition adverse to quality [P.1(c)] (Section 1R15).

Cornerstone: Barrier Integrity

- Green. The inspectors identified two examples of a noncited violation of 10 CFR Part 50, Criterion III, "Design Control," for the failure of engineering personnel to ensure that the design bases of the refueling machine were adequately translated into specifications, drawings, procedures, or instructions. Specifically, for the first example, between October 27, 2006, and October 25, 2007, the licensee inappropriately changed the facility as described in the Updated Final Safety Analysis Report when a modification to the refueling machine introduced a single failure that could result in a failure of both the underload and overload protection features. This change resulted in more than a minimal increase in the consequences of a malfunction, in that the force limits on a fuel assembly grid strap could be exceeded. For the second example, between initial construction and December 5, 2007, procedures and instructions did not limit the stall torque of the hoist motor for the refueling machine. These issues were entered into the corrective action program as Condition Report/Disposition Requests 3030759 and 3068656.

  The finding is greater than minor because it would become a more significant safety concern if left uncorrected in that refueling equipment malfunctions could result in damaged fuel. Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," was used since the Significance Determination Process methods and tools were not adequate to determine the significance of the finding. This finding affects the barrier integrity cornerstone and is determined to have very low safety significance by NRC management review because it was a deficiency that did not result in the actual degradation of fuel (Section 4OA2).

Cornerstone: Occupational Radiation Safety

- Green. Two examples of a self-revealing noncited violation of Technical Specification 5.7.1 were identified for the failure to control a high radiation area. On February 14, 2007, while preparing to perform a remote inspection and boric acid wash down of Unit 2 Letdown Ion Exchange Vessel CHN-D01A, a worker received a dose rate alarm of 141 mr/hr on his electronic dosimeter when he removed the shielded plug from the survey/inspection port. On October 24, 2007, while decontaminating safety injection Tank 2A outlet valve to Loop 2A Valve SIB-UV-614 using a vacuum in the Unit 3 containment, two workers received separate electronic dosimeter alarms of 81 mr/hr and 123 mr/hr approximately 20 minutes apart. The issues were entered into the corrective action program as Condition Report/Disposition Requests 2970612 and 3081978.

  This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the failure to post and control a high radiation area had the potential to increase personnel dose. This occurrence involved individual workers' unplanned, unintended dose that resulted from actions or conditions contrary to licensee procedures, radiation work permit, and Technical Specifications, therefore, this finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Both examples have a crosscutting aspect in the area of human performance associated with work control because work planning did not appropriately plan work activities by incorporating risk insights and job site conditions [H.3(a)] (Section 2OS1).

- Green. The inspectors identified a noncited violation of 10 CFR 20.1501(a) because the licensee failed to completely evaluate the radiological hazard associated with the decontamination of the temporary reactor head. This failure led to the internal exposure of two workers and personnel contamination of two other nearby individuals. The original apparent cause evaluation determined that the radiation protection technicians' decision not to rinse the underside of the temporary reactor head caused the uptakes and contaminations. Upon NRC documentation review and interviews with staff, the licensee determined that the total effective dose equivalent ALARA evaluation of the radiological conditions and the use of appropriate protective equipment did not fully consider the job site conditions or process of decontamination of the temporary reactor head. The issue was entered into the corrective action program as Condition Report/Disposition Request 3046953.

  This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the failure to evaluate the radiological conditions had the potential to increase personnel dose. This occurrence involved individual worker unplanned, unintended dose that resulted from actions or conditions contrary to licensee procedures, radiation work permit, and Technical Specifications, therefore, this finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding has a crosscutting aspect in the area of human performance associated with work control because the work planning did not consider possible risk insights and job sight conditions [H.3(a)] (Section 2OS1).

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a for the failure to follow radiation exposure permit instructions. Specifically, while touring the Unit 3 containment on October 23, 2007, the inspectors questioned six individuals at the pressurizer cubicle on the 120' level. The individuals stated they left their job site and proceeded to a new job site without informing radiation protection and receiving a radiological brief of the conditions at the new job site. The workers were coached by the licensee and the issue was entered into the corrective action program as Palo Verde Action Request 3081935.

This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the noncompliance to a radiation exposure permit instructions had the potential to increase personnel dose. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding has a crosscutting aspect in the area of human performance associated with work practices because the workers did not use human error prevention techniques such as adequate self and peer checking to appropriately evaluate work conditions [(H.4(a)] (Section 2OS1).

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. The violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full power until October 22, 2007, when the unit was shutdown to repair the auxiliary feedwater system (AFW) Train A steam supply valves. The unit returned to full power on November 5. On November 23, the unit was shutdown to repair the Train A balance of plant engineered safety feature actuating system (BOP ESFAS) load sequencer. The unit returned to full power on December 1. On December 6, power was reduced to approximately 73 percent to repair a fault in the Train A heater drain system. The unit returned to full power on December 8 and remained there for the duration of the inspection period.

Unit 2 operated at full power until October 6, 2007, when the unit was manually tripped due to high sodium levels in secondary water systems. The unit returned to full power on October 14 and remained there for the duration of the inspection period.

Unit 3 was shut down for the entire inspection period for a refueling and steam generator (SG) replacement outage.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

Readiness for Seasonal Susceptibilities

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

- December 3, 2007, Unit 1, essential spray pond system Train A
- December 3, 2007, Unit 2, essential spray pond system Trains A and B
- December 3, 2007, Unit 1, essential cooling water system Trains A and B
- December 4, 2007, Unit 3, nuclear cooling water temporary modification

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial Walkdown

The inspectors: (1) walked down portions of the three below listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walk down to the licensee's UFSAR and corrected action program (CAP) to ensure problems were being identified and corrected.

- October 17, 2007, Unit 3, essential cooling water and pool cooling Train B while Train A was out of service
- December 4-5, 2007, Unit 1, AFW system Train A
- December 20, 2007, Unit 3, shutdown cooling (SDC) system Train B

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Quarterly Inspection

The inspectors walked down the six below listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the UFSAR to determine if the licensee identified and corrected fire protection problems.

- October 10-11, 2007, Unit 3, containment, 80-foot, 100-foot, 120-foot, and 140-foot elevations
- November 7, 2007, Unit 1, fire pump house, 100-foot elevation
- November 7, 2007, station black out gas turbine generators
- November 10, 2007, Unit 3, condensate storage pump house and tunnel
- December 3, 2007, Unit 1, main steam (MS) support structure, 80-foot, 100-foot, 120-foot, and 140-foot elevations
- December 6, 2007, Unit 2, MS support structure, 80-foot, 100-foot, 120-foot, and 140-foot elevations

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

Semi-annual Internal Flooding

The inspectors: (1) reviewed the UFSAR, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; (2) reviewed the UFSAR and CAP to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) walked down the five below listed areas to verify the adequacy of: (a) equipment seals located below the floodline,

(b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- December 6, 2007, Unit 2, control building, 74-foot, 100-foot, 120-foot, 140-foot, and 160-foot elevations

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

02.01 Inspection Activities Other Than SG Tube Inspection, PWR Vessel Upper Head Penetration (VUHP) Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination (NDE) activities and, if performed, one to three welds on the reactor coolant system (RCS) pressure boundary. Also review one or two examinations with recordable indications that have been accepted by the licensee for continued service.

The inspectors directly observed the following NDEs:

System	Component/Weld ID	Exam Type
AFW	62-12	ultrasonic test (UT)/dye penetrant test (PT)
AFW	62-13	UT/PT
AFW	13-AF-018-H-002	Visual Test Level 3
AFW	13-AF-018-H-003	Visual Test 3

The inspectors reviewed records for the following NDEs:

System	Component/Weld ID	Exam Type

SG1 Hot Leg	1RCS1	radiography test (RT)
High Pressure Safety Injection (HPSI) Line (3")	1SI14	RT
RCS Cold Leg	1RCS4	RT
MS	2MS2	RT
Safety Injection (SI) (16")	1SI3	RT
MS	1MS8	RT
Feedwater (FW)	2FW8	RT
FW	2FW4	RT
SG2 RCS Cold Leg "B"	2RCS5	RT
SG2 RCS Cold Leg "A"	2RCS3	RT
Downcomer Blowdown	2DB14 (Elbow-Pipe)	RT
Downcomer Blowdown	2DB24 (Elbow-Elbow)	RT
Downcomer Blowdown	2DB1R2 (Nozzle-Reducer)	RT
SG1 RCS Cold Leg "B"	1RCS5 (Elbow-Pipe)	RT
SG1 RCS Cold Leg "A"	1RCS3 (Elbow-Pipe)	RT

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and applicable procedures. The qualifications of all NDE technicians performing the inspections were verified to be current.

The inspectors reviewed records for one NDE examination with a recordable indication that was accepted by the licensee for continued service as listed below. The licensee's acceptance was in accordance with ASME code requirements.

System/Component	Weld ID/Comp ID	Exam Type	Result
SDC Heat Exchanger	75-79	UT	Indication Acceptable

Records from 16 examples of welding on the RCS pressure boundary were examined as follows:

System	Component/Weld Identification
SI	3PSIBV1026/2866117-1, 2, 3, 5, and 6
SG1 Hot Leg	1RCS1
HPSI Line (3")	1SI14
RCS Cold Leg	1RCS4
MS	2MS2
SI (16")	1SI3
MS	1MS8
FW	2FW8
FW	2FW4
SG2 RCS Cold Leg "B"	2RCS5
SG2 RCS Cold Leg "A"	2RCS3
Downcomer Blowdown	2DB14 (Elbow-Pipe)
Downcomer Blowdown	2DB24 (Elbow-Elbow)
Downcomer Blowdown	2DB1R2 (Nozzle-Reducer)
SG1 RCS Cold Leg "B"	1RCS5 (Elbow-Pipe)
SG1 RCS Cold Leg "A"	1RCS3 (Elbow-Pipe)

Welding procedures and NDE of the welding repair conformed to ASME Code requirements and licensee requirements.

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance withe ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the gas tungsten arc welding process (machine and manual) and the shielded metal arc welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

The inspectors completed one sample under Section 02.01.

b. Findings

No findings of significance were identified.

02.02 VUHP Inspection Activities

a. Inspection Scope

The licensee performed NDE of 100 percent of reactor VUHPs. The inspector directly observed a sample of the examinations as listed below:

System	Component/Weld Identification	Examination Method
Control Element Drive Mechanism (CEDM)	CEDM 12	VT2
CEDM	CEDM 17	VT2
CEDM	CEDM 28	VT2
CEDM	CEDM 45	VT2
CEDM	CEDM 62	VT2
CEDM	CEDM 64	UT, eddy current test (ET)
CEDM	CEDM 84	UT, ET
CEDM	CEDM 86	UT, ET

The inspectors reviewed the following sample of examinations in which defects were detected and accepted for continued service using stored electronic data or review of printed records:

System	Examination Method	Result
CEDM 30	ET, UT, PT	Indications determined to be not primary water stress corrosion cracking (PWSCC)
CEDM 39	ET, UT, PT	Indications determined to be not PWSCC
CEDM 50	ET, UT, PT	Indications determined to be not PWSCC

The NDE inspections were performed in accordance with the requirements of NRC Order EA-03-009.

The inspectors completed one sample under Section 02.02.

b. Findings

No findings of significance were identified.

02.03 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

Resident inspectors observed a sample of boric acid corrosion control activities and verified that visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components.

The ISI inspector reviewed three instances where boric acid deposits were found on RCS system piping components:

Component Number	Description	Action Request
3PSIAV105	Containment Spray (CS) Pump SIA-P03 Norm Suction Valve	3005292
3PSIEV214	Safety Injection Tank (SIT) “2A” Sample Isolation Valve	2934501
3PRCNV715	Reactor Coolant Pump (RCP) “2B” Pump Casing Vent Valve	2890601

The condition of the all the components was appropriately entered into the licensee’s CAP, and corrective actions taken were consistent with ASME code requirements. No engineering evaluations were required for any of the instances where leaks were identified during walkdowns.

The inspectors completed one sample under Section 02.03.

b. Findings

No findings of significance were identified.

02.04 SG Tube Inspection Activities

a. Inspection Scope

Unit 3 replaced SGs during this outage and SG tubes were not inspected.

b. Findings

No findings of significance were identified.

02.05 Identification and Resolution of Problems

a. Inspection scope.

The inspection procedure requires review of a sample of problems associated with in-service inspections documented by the licensee in the CAP for appropriateness of the corrective actions.

The inspector reviewed nine corrective action reports which dealt with in-service inspection activities and found the corrective actions were appropriate. Action requests reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee has an appropriate threshold for entering ISI issues into the CAP and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Annual Inspection

a. Inspection Scope

The inspectors reviewed the annual operating examination test results for 2007. Since this was the first half of the biennial requalification cycle, the licensee was not required to administer a written examination. These results were assessed to determine if they were consistent with NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," guidance and Manual Chapter 0609, "Significance Determination Process," Appendix I, "Operator Requalification Human Performance Significance Determination Process," requirements. This review included the test results for a total of 20 crews composed of 91 licensed operators, which included: shift-standing senior operators, staff senior operators, shift-standing reactor operators, and staff reactor operators. There was one crew failure and four individual failures on the simulator scenario portion of the test. There were no individual failures on the job performance measure portion of the test. All failures were remediated following the examination.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

Quarterly Inspection

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The below listed evaluated scenarios were observed. The training scenarios involved: (1) a loss of offsite power and station blackout; and (2) an inadvertent AFW system actuation, anticipated transient without a scram, and loss of coolant accident.

- October 16, 2007, Scenario SES-0-08-F-00, “LOP/LOOP/Blackout”
- October 16, 2007, Scenario SES-0-03-T-00, “Inadvertent AFAS/ATWS/LOCA”

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the two below listed maintenance activities to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, Appendix B of 10 CFR Part 50, and the TSs.

- August 18, 2007, Unit 2, failure of Quality Safety Parameter Display System (QSPDS), heated junction thermocouples, and core exit temperature thermocouples, resulting in inoperability of QSPDS Train A

- October 17 - 18, 2007, Unit 1, steam supply to AFW Pump A bypass Valve SGA-UV-138A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

Introduction: A self-revealing Green noncited (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” occurred as a result of engineering personnel’s inadequate corrective actions following the September 17, 2007, failure of steam supply to AFW Pump A bypass Valve SGA-UV-138A.

Description: On September 17, 2007, Valve SGA-UV-138A failed to open as required during the performance of the quarterly surveillance test, per Procedure 73ST-9AF02, “AFA-P01 Inservice Test,” Revision 39. Valve SGA-UV-138A is a 1-inch, stainless steel, single pilot assisted, 125 Vdc solenoid operated valve that provides the initial motive force to AFW Pump A. This valve opens to accelerate the turbine from rest to approximately 900 rpm prior to steam supply Valve SGA-UV-138 opening. This arrangement precludes the turbine from tripping on over speed during startup and functions to prime and lubricate the hydraulic control subsystems. This valve set supplies steam from SG 2. A redundant set of valves (SGA-UV-134A and SGA-UV-134) provide steam from SG 1. Either set of valves is capable of supplying sufficient steam to fulfill the AFW system’s safety function. Both sets of valves receive power from Class 125 Vdc Bus PKA-M41.

The valve failure was initially entered into the CAP on September 17, 2007, as Palo Verde Action Request (PVAR) 3064151. Significant Condition Report/Disposition Request (CRDR) 3064675 was written on September 18. An investigation conducted by engineering personnel discovered that the valve internal assembly was mechanically bound inside the bonnet due to the presence of an iron oxide coating on the plunger. The foreign material was evaluated and found to be primarily iron based material with trace amounts of chromium, nickel, and other material. Although multiple sources of the material were postulated, no definitive conclusion was reached as to the actual source. Work on an upstream valve during Refueling Outage 1R13 was attributed as the most probable source for the material. On September 18, the valve internals were replaced and the valve was reassembled. The valve was retested on September 19. Later on September 19, both Valves SGA-UV-134A and SGA-UV-138A were disassembled and inspected and identified no concerns. Following this inspection, both valves were reassembled and retested on September 20. On September 21, 2007, following all postmaintenance and surveillance testing, the AFW steam supplies were declared operable.

As documented in the prompt operability determination (OD) dated September 28, 2007, Revision 0, the licensee considered the failure a one time occurrence isolated to Valve SGA-UV-138A that had been corrected. In addition, it was expected that the condition would not recur absent maintenance activities that involve breaching the valve or piping. However, since the root cause of the failure had not yet been determined, a potential degraded condition existed with possible transportability to both bypass steam supply valves for the turbine driven AFW pump. Consequently, a corrective action was implemented to increase the frequency of stroking Valve SGA-UV-138A from quarterly to monthly. Completion of the investigation for Significant CRDR 3064675 was expected to establish a basis for returning to normal testing.

On October 15, 2007, Valve SGA-UV-138A failed to close while performing the increased frequency testing. The valve was disassembled and the internals were removed and inspected. An iron oxide coating was observed on the plunger. An analysis of the material determined that it was the same consistency as the sample analyzed from the September 17, 2007, failure. In both events, accumulation of iron oxide on the internal plunger created frictional forces between the plunger and the interior surfaces of the bonnet. On October 22, 2007, the unit was shutdown when the TS allowed outage time expired.

The source of the debris was not definitively determined during the investigation following the October 15, 2007, failure. However, additional corrective actions were implemented to preclude recurrence. Inspections were performed on both Valves SGA-UV-134A and SGA-UV-138A, downstream check Valve SGE-V-888, steam trap isolation Valve SGB-UV-1134, and all associated piping. The inspections did not identify the presence of foreign material. Additionally, the valve body for Valve SGA-UV-138A was dye penetrant tested during this inspection with no abnormalities observed. The slope of the 1-inch steam line upstream of Valve SGA-UV-138A was determined to be nonconforming with the design, in that, the piping sloped toward the valve instead of away from the valve. The piping upstream of Valve SGA-UV-138A was reworked to correct the slope. The slope of the same line in Unit 2 was evaluated and determined to be in conformance with design. The slope of the same line in Unit 3 was evaluated and determined to be nonconforming with the design. The piping in Unit 3 was reworked during Refueling Outage 3R13.

A flushing plan was developed and implemented for the piping associated with Valve SGA-UV-138A. The line was flushed three times and the valve internals inspected after each flush. The first flush produced a small amount of oxide residue on the valve's internal components. The second flush produced much less residue and a third flush produced no residue on the valve's internal components. Valve SGA-UV-138A was then reassembled with new internals and retested and returned to service on October 29, 2007. Since the return to service, Valve SGA-UV-138A has been successfully stroke tested eight times. In addition, it has been disassembled and inspected on November 4, November 27, and on December 27, 2007. During an inspection on November 27, 2007, a slight residue of red dust was observed on the valve's internal components. The dust was easily wiped off. There was no other evidence of foreign material on the valve's internal components during these inspections. Future corrective actions will be based on stroke test and inspection results.

<u>Analysis</u>: The performance deficiency associated with this finding was the failure of engineering personnel to implement adequate corrective actions following the September 17, 2007, valve failure. The finding is greater than minor because a failure to open is associated with the equipment performance attribute of the mitigating systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, a failure to close is associated with the SSC and barrier performance attribute of the barrier integrity cornerstone and affects the associated cornerstone objective to provide reasonable assurance that physical design

barriers protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, this finding is determined to have very low safety significance because the finding did not result in a loss of safety function under the mitigating systems cornerstone and did not result in an actual open pathway in the physical integrity of the reactor containment under the containment barrier cornerstone. This finding has a crosscutting aspect in the area of human performance associated with work control because the facility did not dedicate the manpower and expertise necessary to coordinate work activities to incorporate actions to support long term equipment reliability and safety system availability [H.3(b)].

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. Contrary to the above, between September 17 and October 15, 2007, measures to assure that the cause of the failure of steam supply to AFW Pump A bypass Valve SGA-UV-138A was determined and corrective actions taken to preclude repetition of a significant condition adverse to quality were inadequate. Specifically, on September 17, 2007, Valve SGA-UV-138A failed to open as required during performance of the quarterly surveillance test. The cause of the failure was determined to be foreign material on the valve's internal components. Corrective actions were implemented but the source of the debris was not definitively identified. Subsequently, on October 15, 2007, the valve failed to close. Subsequent investigation indicated that the failure was caused by foreign material on the valve's internal components. Because this finding is of very low safety significance and has been entered into the licensee's CAP as CRDR 3078032, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000528/2007005-01, "Failure to Take Adequate Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality."

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

a. Inspection Scope

Risk Assessment and Management of Risk

The inspectors reviewed the below listed assessment activity to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- December 5, 2007, Unit 1, risk assessment and management during scheduled emergency diesel generator (EDG) Train B outage

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

Emergent Work Control

The inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the UFSAR to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- September 18 - 28, 2007, Unit 1, troubleshooting efforts associated with the steam supply to AFW Pump A bypass Valve SGA-UV-138A

- October 15, 2007, Unit 1, troubleshooting efforts associated with the steam supply to AFW Pump A bypass Valve SGA-UV-138A as described in corrective maintenance work order (WO) 3076750

- November 13 - 14, 2007, Unit 1, EDG Train B fuel injection pump replacement

- November 26, 2007, Unit 1, BOP ESFAS sequencer Train A troubleshooting and repair

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. <u>Findings</u>

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and night orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the UFSAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- September 12, 2007, Unit 2, EDG turbocharger lube oil (LO) filter clogging as described in PVAR 3061932

- October 30, 2007, Unit 1, steam leak from bonnet of manual isolation Valve SGE-V886 for the steam supply to AFW Pump A bypass Valve SGA-UV-138A as described in PVAR 3084439

- December 14, 2007, Unit 3, impact of desicant dropped in reactor vessel to core reload and shutdown operations as described in deficiency WO 3107411

- December 14, 2007, Unit 2, operability assessment associated with shutdown cooling suction isolation Valve SIA-UV-651 vibration as described in PVAR 3109083

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

Introduction: The inspectors identified two examples of a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the failure of operations personnel to follow procedures and adequately evaluate degraded and nonconforming conditions to support operability decision-making.

Description: The first example occurred on September 12, 2007, when the Unit 2 EDG Train B was started for a surveillance test per Procedure 40DP-9OP08, "Diesel Generator Test Record," Revision 44. The auxiliary operator noted that the pressure for the turbocharger LO indicated low in the normal band, and informed the shift manager (SM) of the readings. The SM and shift technical advisor reviewed log readings for the turbocharger LO pressure for the period including January 2006 through September 12, 2007, and noted a lowering trend. Based on the lowering trend, the standby filter in the duplex filter arrangement was selected. The turbocharger LO pressure indication increased when the standby filter was selected. PVAR 3061932 was initiated to investigate the lowering turbocharger LO pressure indication. However, the OD process was not entered since the SM determined that a degraded condition did not exist once the standby filter was selected. Additionally, the SM believed that the clogged filter only constituted a material condition since filters can be swapped while the EDG is running. The condition associated with the cause of the filter clogging was not considered in the initial operability assessment as a condition that could impact operability of the EDG.

The inspectors questioned the SM's decision-making to determine that the OD process as described in Procedure 40DP-9OP26, “Operability Determination and Functional Assessment,” Revision 18, was not applicable to the lowering turbocharger LO pressure condition. The inspectors expressed concern that a degraded condition may exist that called into question the ability of the EDG to perform its safety function, in that, the cause of the filter clogging had not been identified and corrected. The inspectors observed that operations personnel failed to consider both the degrading filter condition and the degrading/erratic oil pressure indication as conditions that could challenge the ability of the EDG to perform its safety function. On October 5, 2007, PVAR 3061932 documented a prompt OD for the EDG Train B degrading turbocharger LO pressure condition. The prompt OD concluded that there was an adequate basis to support a reasonable expectation of operability for EDG Train B since the apparent filter fouling rate, and the ability to select the standby filter while the EDG was running, was such that the EDG could function for its entire 7-day mission time. The licensee initiated CRDR 3068929 to evaluate and correct the failure to perform an OD when the lowering LO pressure condition was initially identified.

On October 11, 2007, the prompt OD was revised to incorporate information developed during the CRDR investigation. The extent of condition review identified a declining trend for the pressure indication for turbocharger LO on Unit 2 EDG Train A. The review determined that if the degradation continued to progress linearly, the ability of the EDG to perform its safety function to operate for the mission time could be challenged. The revision to the prompt OD concluded that there was an adequate basis to support a reasonable expectation of operability for EDG Train A since the calculated filter fouling rate, and the ability to select the standby filter while the EDG was running, was such that the EDG could function for its entire 7-day mission time. The apparent cause evaluation documented in CRDR 3062419 determined that the lowering turbocharger LO pressure trend was attributed to filter plugging. The filter plugging occurred, in part, since the wrong filter size rating for the turbocharger bearings were installed. The installed filter elements had a “nominal rating” of 5 microns, whereas, the vendor technical manual states that the filters were 10 microns. It appears that the installed turbocharger filters were too fine a rating for the turbocharger bearing application. Additionally, the licensee determined that the preventive maintenance associated with changing the filters and LO was not at the optimal frequency. The licensee has initiated actions to replace the installed 5 micron filters with filters of a nominal rating of 10 microns and to optimize the preventive maintenance frequency.

The second example occurred on October 29, 2007, at approximately 5:15 a.m., when a body to bonnet steam leak was observed by the inspectors on Valve SGE-V886. This valve is the sole upstream manual isolation for Valve SGA-UV-138A. The inspectors were informed by site personnel that the control room was aware of the leak. At this time, the steam leak had not been entered into the CAP. Control room shift personnel stated that the steam leak was a material condition and not a degraded condition. Consequently, the licensee did not view the issue as an operability concern. This position was documented in an Operations Decision Making Issue/Emergent Condition Checklist dated October 29, 2007.

At approximately 9 a.m. on October 29, 2007, the inspectors observed that the characteristics of the leak had changed and appeared to be getting worse. In addition, the inspectors noted that the direction and size of the plume had changed, causing a concern with the impact to nearby equipment. The inspectors raised additional questions regarding AFW operability and containment isolation with operations shift personnel and operations management. At 12:10 p.m. on October 29, 2007, the issue was entered into the CAP, however, the steam leak was still not considered an operability issue.

Operations Department Practices 16, Revision 8, states, in part, that if the potential exists to impact a TS SSC, then perform an operability evaluation per the OD procedure. Procedure 40DP-9OP26 required the OD process be entered when a TS SSC or TS Support SSC safety function is called into question. The immediate OD is typically performed in the order of 2 hours. A more rigorous prompt OD is typically performed in the order of 24 hours.

In the afternoon of October 29, 2007, the inspectors continued to ask additional questions regarding operability of fire suppression, containment isolation, and AFW systems. On October 30, 2007, an immediate OD was performed and approved by the SM at 5:54 a.m. The OD addressed operability of the steam driven AFW pump, environmental qualifications of equipment in the room, fire suppression system operability, room habitability, and containment isolation issues. The prompt OD was subsequently performed on October 31, 2007, and provided additional operability justification details. On November 1, 2007, seal weld repairs were completed to isolate the steam leak.

Analysis: The failure to adequately implement the OD process was a performance deficiency. The finding is greater than minor because the degraded turbocharger LO filter is associated with the equipment performance attribute of the mitigating systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, the steam leak on manual isolation Valve SGE-V886 is associated with the SSC and barrier performance attribute of the barrier integrity cornerstone and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheets, the finding is determined to have very low safety significance because the finding did not result in a loss of safety function under the mitigating systems cornerstone and did not result in an actual open pathway in the physical integrity of the reactor containment under the containment barriers cornerstone. The example of this finding related to lowering turbocharger lube oil pressure has a crosscutting aspect in the area of human performance associated with decision-making because the licensee did not use conservative assumptions for operability decision-making when evaluating degraded and nonconforming conditions [H.1(b)]. The example of this finding related to the body to bonnet steam leak has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not properly classify, and thoroughly evaluate the operability for a condition adverse to quality [P.1(c)].

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires that activities affecting quality shall be prescribed by instructions, procedures, or drawings, and shall be accomplished in accordance with those instructions, procedures, and drawings. The assessment of operability of safety-related equipment needed to mitigate accidents was an activity affecting quality and was implemented by Procedure 40DP-9OP26, "Operability Determination and Functional Assessment," Revision 18. Procedure 40DP-9OP26, Step 3.1.1, requires entry into the OD process upon discovery of circumstances where the operability of any SSC described in TSs is called into question upon discovery of degraded conditions. Contrary to the above, between September 12 and October 5, 2007, and between October 29 and October 30, 2007, operations personnel failed to enter the OD process upon discovery of degraded conditions that called into question the operability of a SSCs described in TSs. Specifically, operations personnel failed to adequately evaluate the operability of the Unit 2 EDG Train B during a lowering turbocharger LO pressure condition and the Unit 1 AFW system Train A steam supply during a body to bonnet steam leak on Valve SGE-V886. Because the finding is of very low safety significance and has been entered into the CAP as CRDR 3068929 and PVAR 3084439, this violation is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy: NCV 05000528; 05000529/2007005-02, "Two Examples of a Failure to Properly Implement the Operability Determination Process."

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Annual Review

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

- October 15 - December 12, 2007, Unit 3, relocation of Valve SIA-UV-651 per design modification WO 2914420

- October 24, 2007, Unit 1, installation of manual isolation valves downstream Valves SGA-UV-134A and SGA-UV-138A per design modification WO 3054112

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the five below listed post maintenance test activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the UFSAR to determine if the licensee identified and corrected problems related to postmaintenance testing.

- October 19, 2007, Unit 1, AFW system steam supply pipe flush per Procedure 40OP-9AF01, “Essential Auxiliary Feedwater System,” Revision 38

- November 14, 2007, Unit 1, EDG Train B fuel injection pump replacement

- November 26, 2007, Unit 1, BOP ESFAS sequencer Train A

- December 3, 2007, Unit 3, vortex test following relocation of Valve SIA-UV-651 per Procedure 40TI-9ZZ07, “Shutdown Cooling Vortex Test,” Revision 5

- December 12-13, 2007, Unit 3, recalibration of refueling water (vessel) level indication system following relocation of Valve SIA-UV-651, per Procedure 32MT-3RC03, “Refueling Water Level Indicating System Instrument Calibration - Train A,” Revision 15

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

Unit 2 Forced Outage for High Sodium In Secondary Water Systems

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, “Loss of Decay Heat Removal”: (1) the risk control plan, (2) decay heat removal, (3) reactivity control, (4) heatup and coldown activities, (5) restart activities, and (6) licensee identification and implementation of appropriate corrective actions associated with outage activities.

Unit 1 Forced Outage for Valve SGA-UV-138A Failure

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, “Loss of Decay Heat Removal”: (1) the risk control plan, (2) tagging/clearance activities, (3) decay heat removal, (4) reactivity control, (5) containment closure, (6) heatup and coldown activities, (7) restart activities, and (8) licensee identification and implementation of appropriate corrective actions associated with outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris; and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging.

Unit 1 Forced Outage for BOP ESFAS Train A Sequencer Failure

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, “Loss of Decay Heat Removal”: (1) the risk control plan, (2) decay heat removal, (3) reactivity control, (4) containment closure, (5) heatup and coldown activities, (6) restart activities, and (7) licensee identification and implementation of appropriate corrective actions associated with outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris; and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and TSs to ensure that the two below listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method to demonstrate TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator (PI) data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms set points. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- October 5, 2007, Unit 3, local leak rate testing of containment Penetration 42B per Section 8.21 of Procedure 73ST-9CL01, “Containment Leakage Type 'B' and 'C' Testing,” Revision 30

- October 29, 2007, Unit 1, inservice test of turbine driven AFW pump per Procedure 73ST-9AF02, “AFA-P01 Inservice Test,” Revision 39

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the UFSAR, plant drawings, procedure requirements, and TSs to ensure that the two below listed temporary modifications were properly implemented. The inspectors: (1) verified that the modifications did not have an effect on system operability/availability, (2) verified that the installation was consistent with modification documents, (3) ensured that the postinstallation test results were satisfactory and that the impact of the temporary modifications on permanently installed SSCs were supported by the test, (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings, and (5) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with temporary modifications.

- October 12 - 18, 2007, Unit 3, temporary alternate cooling to nuclear cooling water Train B via Procedures 31MT-9WP02, “Installation and Removal of Temporary Cooling Towers to NC Heat Exchanger for PW System Outage,” Revision 7, and 40OP-9PW01, “Plant Cooling Water,” Revision 29

- October 15, 2007, Unit 3, temporary power installation for support of Class 1E Bus E-PBA-SO3

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

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

- PI events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personal dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Radiation Exposure Permits, procedures, engineering controls, and air sampler locations
- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools.
- Self-assessments, audits, licensee event reports (LERs), and special reports related to the access control program since the last inspection (Sample #8)
- Corrective action documents related to access controls
- Radiation Exposure Permit briefings and worker instructions
- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

The inspectors completed 20 of the required 21 samples.

b. Findings

.1 Failure to Post and Control High Radiation Areas

Introduction: The inspectors identified two examples of a Green self-revealing NCV of TS 5.7.1 resulting from radiation protection personnel failures to control high radiation areas.

Description: The first example occurred on February 14, 2007. While preparing to perform a remote inspection and boric acid wash down of Unit 2 Letdown Ion Exchange Vessel CHN-D01A, a worker received a dose rate alarm on his electronic dosimeter when he removed the shielded plug from the survey/inspection port. The radiation protection technician covering the job had handed the worker the tool to remove the plug from a survey port and turned his attention to preparing for an air sample. The worker used the tool and removed the plug without the radiation protection technician's knowledge and received a dose rate alarm of 141 mr/hr. The individuals were working on a radiation exposure permit which did not allow access to a high radiation area and the dose rate alarm setpoint was 75 mr/hr. All work was stopped and the plug reinstalled.

Prior to the start of the job, the radiation protection section leader conducting the job brief was unaware of the current ion exchanger status (i.e., recent usage and dose rates) and minimal discussion on risk insights and contingency plans was held. Also, the removal tool for the survey port was approximately 6 inches in length placing the worker in close proximity to streaming radiation. For corrective actions, the individuals involved were coached about proper planning, controls, and communications during work activities. Also, the licensee initiated a corrective action to evaluate producing a longer tool handle to move personnel further away from the plane of the survey port.

The second example occurred on October 24, 2007. While decontaminating SI Tank 2A outlet valve to Loop 2A Valve SIB-UV-614 using a vacuum, two workers received electronic dosimeter alarms. The first worker received a dose rate alarm of 81 mr/hr, but did not hear it due to noise from the vacuum, and did not take the required actions upon receiving an alarm. Work continued for approximately 20 minutes and a second worker received a dose rate alarm of 123 mr/hr. The second alarm was heard and work was stopped and put into a safe condition. The radiation protection technician covering the job performed a survey and found that the vacuum hose had a dose rate reading of 150 mr/hr at 30 centimeters, constituting a high radiation area. All the individuals were working on a radiation exposure permit which did not allow access to a high radiation area and the dose rate alarm setpoint was 75 mr/hr.

Analysis: The failure to post and control high radiation areas was a performance deficiency. This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the failure to post and control a high radiation area had the potential to increase personnel dose. This occurrence involved individual workers' unplanned, unintended dose that resulted from actions or conditions contrary to licensee procedures, radiation work permit, and TSs; therefore, this finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable (ALARA) planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Both examples have crosscutting aspects in the area of human performance associated with work control component because the work planning did not appropriately plan work activities by incorporating risk insights and job site conditions [H.3(a)].

The finding was self-revealing because the licensee was made aware of both high radiation area conditions following an individuals' electronic dosimeter alarm.

Enforcement: TS 5.7.1 requires, in part, that the licensee barricade and conspicuously post high radiation areas in lieu of the requirement of 10 CFR 20.1601(a). Pursuant to 10 CFR 20.1003, "high radiation area" means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates. Contrary to the above, on February 14, 2007, and on October 24, 2007, the licensee failed to adequately post and control access to high radiation areas resulting in three workers receiving electronic dosimeter high dose rate alarms. Because this violation was of very low safety significance and has been entered into the licensee's CAP as CRDRs 2970612 and 3081978, it is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000529/2007005-03, "Two Examples of a Failure to Post and Control High Radiation Areas."

.2 Failure to Evaluate the Radiological Hazard Caused From Decontamination

Introduction: The inspectors identified a self-revealing Green NCV of 10 CFR 20.1501(a) because radiation protection personnel failed to completely evaluate the radiological hazard of decontaminating the temporary reactor head, leading to internal exposure of two workers and personnel contamination of two other nearby individuals.

Description: During Refueling Outage 1R13, the temporary reactor head was to be moved to the top of the pressurizer cubicle for decontamination. The prejob brief included instructions that the temporary reactor head should be completely rinsed including the underside. The temporary reactor head was lifted from the refuel pool and the top rinsed. The temporary reactor head was moved to the north side of the cavity where maintenance personnel removed services and rigging, and the radiation protection technician performed surveys which included one smear of the underside of the temporary reactor head indicating 1,000,000 dpm/100cm^2. The radiation protection technician decided the underside did not need to be rinsed even though the prejob brief included rinsing the underneath of the temporary reactor head. The temporary reactor head was moved to the pressurizer cubicle where it sat and dried longer than normal prior to decontamination. Subsequently, two workers received an internal exposure of 9 mrem and 5 mrem, and two other workers were contaminated.

The draft apparent cause evaluation determined that the radiation protection technicians' decision not to rinse the underside of the temporary reactor head was the apparent cause with contributing factors such as using somewhat volatile chemicals during decontamination, containment ventilation being secured, and the temporary reactor head being allowed to dry longer than normal. Based on further NRC documentation review and interviews with staff, the licensee subsequently determined that their total effective dose equivalent ALARA evaluation of the radiological conditions and the use of appropriate protective equipment did not fully consider the job site conditions or process of decontaminating the temporary reactor head. The licensee reopened the corrective action document and was re-evaluating the apparent cause and corrective action for this occurrence.

Analysis: This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that not completely evaluating the radiological conditions had the potential to increase personnel dose. This occurrence involved individual worker unplanned, unintended dose that resulted from actions or conditions contrary to licensee procedures, radiation work permit, and TSs; therefore, this finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The inspector determined that this finding was of very low safety significance because it did not involve: (1) an ALARA planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding has a crosscutting aspect in the area of human performance associated with work control because the work planning did not consider possible risk insights and job sight conditions [H.3(a)].

The finding is self-revealing because the licensee became aware of the condition when workers personnel contamination monitors alarmed.

Enforcement: 10 CFR 20.1501(a), requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. Pursuant to 10 CFR 20.1003, a “survey” means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. 10 CFR 20.1201(a), states, in part, that the licensee shall control the occupational dose to individual adults to specified limits. Contrary to the above, on June 14, 2007, radiation protection personnel failed to completely evaluate the radiological hazard of decontamination of the temporary reactor head. Consequently, two workers were internally exposed and two additional workers were contaminated. Because the violation is of very low safety significance and has been entered into the licensee's CAP as CRDR 3046953, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000528/2007005-04, "Failure to Evaluate the Radiological Hazard Caused from Decontamination."

.3 Failure to Follow Procedural Guidance and Radiation Work Instructions

Introduction: The inspectors identified a Green NCV of TS 5.4.1.a for the failure to follow procedural guidance and radiation exposure permit instructions.

Description: While touring the Unit 3 containment on October 23, 2007, the inspectors questioned six individuals at the pressurizer cubicle on the 120' level. The individuals stated they entered to perform a job on the 90' level but were redirected by their supervisor to a job at the pressurizer cubicle. The individuals stated they left their job site and proceeded to the new job site without informing radiation protection and receiving a radiological brief or reviewing current survey data of the conditions at the new job site. The lead worker knew the general area radiation levels and stated that he knew the radiological conditions of the area because he installed scaffolding in the area the previous week. The workers were coached by the licensee. Based on questioning of other individuals by the inspectors and the licensee, it was determined that this was an isolated event.

Analysis: The failure to follow procedural guidance and radiation exposure permit instructions is a performance deficiency. This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the noncompliance to a procedure had the potential to increase personnel dose. This occurrence involved individual worker unplanned, unintended dose that resulted from actions or conditions contrary to licensee procedures, radiation work permit, and TSs; therefore, this finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The inspector determined that this finding was of very low safety significance because it did not involve: (1) an ALARA planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding has a crosscutting aspect in the area of human performance associated with work practices because the six individuals did not perform adequate self and peer checking to appropriately evaluate work conditions (H.4(a)).

The finding was NRC identified because the licensee was made aware of the situation through questioning of workers by the NRC inspector.

Enforcement: TS 5.4.1 requires procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends, in Section 7, procedures for access control to radiation areas including a radiation work permit system. Implementing Procedure 75DP-9RP01, "Radiation Exposure and Access Control," Revision 10, Section 3.6.2.1, states, in part, that by signing a radiation exposure permit, individuals have indicated they read and understood the radiation exposure permit requirements and will comply with them. Radiation Exposure Permit 3-3508J, “In Service Inspections and Associated Work,” Task 2, stated, in part, for the individuals to review current radiological survey data for the work area prior to entry. Contrary to the above, on October 23, 2007, six individuals did not review current radiological survey data for the work area prior to entry to their job site. Because this violation was of very low safety significance and has been entered into the licensee's CAP as PVAR 3081935, it is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000530/2007005-05, “Failure to Follow Procedural Guidance and Radiation Work Instructions.”

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspector used the requirements in 10 CFR Part 20 and the licensee’s procedures required by TS as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed:

- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Workers’ use of the low dose waiting areas
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Corrective action documents related to the ALARA program and follow-up activities, such as initial problem identification, characterization, and tracking
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended verses actual work activity doses and the reasons for any inconsistencies
- Person - hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered

- Exposure tracking system

- Exposures of individuals from selected work groups

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Source - term control strategy or justification for not pursuing such exposure reduction initiatives

- Specific sources identified by the licensee for exposure reduction actions and priorities established for these actions, and results achieved against since the last refueling cycle

- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results

The inspector completed 7 of the required 15 samples and 8 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors sampled licensee data for the Mitigating System(s) Performance Index (MSPIs) listed below for the period from April 1, 2006, through September 30, 2007, for Units 1, 2, and 3. The definitions and guidance of Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used to verify the licensee's basis for reporting unavailability and unreliability in order to verify the accuracy of PI data. The inspectors reviewed operating logs, Limiting Condition for Operation logs, CRDRs, and the maintenance rule database to verify that the licensee properly accounted for planned and unplanned unavailability as part of the assessment. The inspectors sampled data to verify that the licensee: (1) accurately documented the actual unavailability hours for the MSPI systems, and (2) accurately documented the actual unreliability information for each MSPI monitored component. In addition, the inspectors interviewed licensee personnel associated with PI data collection and evaluation.

- MSPI - Residual Heat Removal System
- MSPI - Heat Removal System
- MSPI - Cooling Water Systems

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed nine samples.

Cornerstone: Occupational Radiation Safety

Occupational Exposure Control Effectiveness

The inspector reviewed licensee documents from January 1 through November 30, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's TSs), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 5). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. PI definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

Cornerstone: Public Radiation Safety

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

The inspector reviewed licensee documents from January 1 through November 30, 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the PI data. PI definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

The inspectors performed a daily screening of items entered into the licensee's CAP. This assessment was accomplished by reviewing daily summary reports for PVARs, and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the CAP; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

.2 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the four below listed issues for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- September 24 - December 7, 2007 Units 1, 2, and 3, design issues with refueling machine as noted in PVAR 3048775
- October 15 - December 13, 2007, Units 1, 2, and 3, design adequacy of masonry wall in control room as noted in PVAR 3102815
- October 26, 2007, Unit 3, valve technicians opened the wrong unit circuit breakers for HPSI motor-operated valves as described in CRDR 3087295
- December 3, 2007, Unit 1, engineered safety feature pump room exhaust air cleanup system (PREACS) declared inoperable due to propping Door Y-1-06 open when routing a hose through the door for maintenance as noted in PVAR 3103619

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

Introduction: The inspectors identified two examples of a Green NCV of 10 CFR Part 50, Criterion III, “Design Control,” for the failure of engineering personnel to ensure that the design bases of the refueling machine were adequately translated into specifications, drawings, procedures, or instructions.

Description: The first example occurred on October 27, 2006, for Unit 2, and on June 17, 2007, for Unit 1, when the licensee upgraded the respective refueling machines. The upgrade included a new programmable logic controller (PLC) that provided protection during fuel movements. These protection features included, in part, independent underload and overload functions.

The basis for the overload and underload hoist limits described in Combustion Engineering Specification SYS80-486-0880, “General Engineering Specification for Reactor Servicing Equipment,” Revision 5, designates a band of ±180 pounds from the nominal fuel weight. Exceeding the ±180-pound limit could cause grid strap damage and potential damage to the fuel due to fretting wear on the fuel cladding. The overload and underload limits are set to a band of ±150 pounds from the nominal fuel weight to ensure that the ±180 pound limits noted in Specification SYS80-486-0880 are not exceeded.

Section 9.1.4 of the UFSAR describes the fuel handling system, including the refueling machine. Section 9.1.4.1.2 states that, “No single interlock failure will result in damaging or dropping of the fuel. Where results are considered possible, redundant switches, mechanical restraints, and physical barriers are employed as well as limiting the hoist stall torque and loading capability to values below those which would result in damage to the fuel.”

On June 19, 2007, the licensee initiated CRDR 3030759 to document a failure of the hoist latch pivot block that occurred during the Unit 1 core reload. The latch limit switch remained actuated which caused the input to the PLC to block the 150-pound underload and overload protection features function. The inspectors noted that a single failure of the hoist latch pivot block ultimately disabled both interlocks. The licensee did not recognize the single failure design requirement concern during the development and installation of the new refueling machines for Units 1 and 2, or during the troubleshooting and repair efforts for the Unit 1 pivot block failure until identified by the inspectors after the Unit 1 core reload was completed.

Following the inspectors' review, the licensee completed an additional assessment and determined no other single failure vulnerabilities existed. Condition Report Action Item (CRAI) 3083132 was initiated on October 25, 2007, to ensure that the PLC logic for the Units 1 and 2 refueling machines was modified to preclude the failure of the 150-pound underload and overload protection features. This action item is scheduled to be completed prior to the next fuel handling activities in Units 1 and 2. The design upgrade for the Unit 3 refueling machine was completed on December 7, 2007, which incorporated the correct design associated with the PLC logic.

The second example occurred during review of the modifications and upgrades to the refueling machine, the inspectors noted that the stall torque of the refueling bridge hoist motor was referenced in Section 9.1 of the UFSAR, Combustion Engineering Specification SYS80-486-0880, “General Engineering Specification for Reactor Servicing Equipment,” Revision 5, and the Fuel Handling Design Basis Manual. The documents stated that the stall torque of the motor was limited to ensure that the fuel assembly tensile loads were not exceeded. This feature was in addition to the underload and overload protection features provided by the refueling machine hoist weight system. UFSAR, Section 9.1.4.1.2, specified that, for the refueling machine hoist overload interlock, no single failure will result in damaging or dropping fuel. Additionally, the hoist stall torque is limited to values below which would result in damage to fuel. Table 9.1-4 of the UFSAR discusses the failure mode analysis of fuel handling equipment, and notes that for a failure of the hoist indicating system, the maximum stall torque of the hoist motors will not damage a fuel bundle.

The inspectors determined that the licensee did not have a record, procedure, or specification that limited the tensile force generated by the hoist motor at stall torque conditions. Because the power supply to the hoist motor is common to the entire refueling bridge, the current drawn by the hoist motor at stall torque conditions was less than the refueling machine supply breaker trip setpoint. Therefore, the tensile force generated by the hoist motor could reach, or exceed, the maximum stall force. This condition was corrected when the licensee installed new refueling bridge controls and hoist motors on all three units.

<u>Analysis</u>: The performance deficiency associated with this finding involved the failure of engineering personnel to translate design requirements into specifications, drawings, procedures, or instructions for the refueling machine interlocks and refueling bridge hoist motor. The finding is greater than minor because it would become a more significant safety concern if left uncorrected in that refueling equipment malfunctions could result in damaged fuel. Manual Chapter 0609, Appendix M, “Significance Determination Process Using Qualitative Criteria,” was used since the Significance Determination Process methods and tools were not adequate to determine the significance of the finding. This finding affects the barrier integrity cornerstone and is determined to have very low safety significance by NRC management review because it was a deficiency that did not result in the actual degradation of fuel.

<u>Enforcement</u>: 10 CFR Part 50, Criterion III, “Design Control,” states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis be translated into specifications, drawings, procedures, and instructions. Contrary to this, between October 27, 2006, and October 25, 2007 (first example), and prior to December 5, 2007 (second example), the licensee did not ensure the design basis was translated into specifications, drawings, procedures, or instructions. Specifically, the new refueling machines for Units 1 and 2 were modified such that a single interlock failure could result in a failure of both the underload and overload protection features. Additionally, procedures and instructions did not limit the stall torque

of the hoist motor for the refueling machine. Because this issue is of low safety significance and has been entered into the licensee's CAP as CRDRs 3030759 and 3068656, this violation is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy: NCV 05000528; 05000529; 05000530/2007005-06, "Two Examples of Inadequate Design Controls for Refueling Machine."

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semi-annual trend review of repetitive or closely related issues that were documented in corrective action documents to identify trends that might indicate the existence of more safety significant issues. The inspectors reviewed corrective action documents for Refueling Outage 3R13 and the Unit 1 SG replacement outage.

- A review of an adverse trend associated with fires during replacement SG Refueling Outage 3R13, as noted in CRDRs 3078329, 3082305, 3083184, 3086462, and 3102290

Documents reviewed by the inspectors are listed in the attachment.

Inspectors completed one sample.

b. Findings

No findings of significance were identified.

.4 Multiple/Repetitive Degraded Cornerstone Column and Crosscutting Issues Followup Activities

The NRC performed the Inspection Procedure 95003 supplemental inspection and held the final exit meeting on December 19, 2007. Results of the inspection will be documented in NRC Inspection Report 05000528; 05000529; and 05000530/2007012. The licensee submitted their performance improvement plan to the NRC on December 31, 2007.

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

Section 1R15.1 describes a finding where operations personnel had an inappropriately high threshold for identifying material conditions as a degraded condition.

The inspector evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)

- ALARA Planning and Controls (Section 2OS2)

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

Event Follow Up

a. Inspection Scope

The inspectors reviewed the five below listed events and degraded conditions for plant status and mitigating actions to: (1) provide input in determining the appropriate agency response in accordance with Management Directive 8.3, "NRC Incident Investigation Program"; (2) evaluate performance of mitigating systems and licensee actions; and (3) confirm that the licensee properly classified the event in accordance with emergency action level procedures and made timely notifications to NRC and state/governments, as required.

- October 16 - 19, 2007, Unit 1, SG 2 steam supply to Valve AFA-P01 bypass Valve SGA-UV-138A failure to open during surveillance testing

- On October 16, 2007, the Unit 3 refueling canal level was noted to slowly increase about 1 inch while spent fuel pool (SFP) level slowly decreased about 3 inches. The levels equalized between the SFP and RCS. The licensee determined that the SFP gate to the transfer canal was leaking slightly. The licensee inspected and reset the SFP gate and did not identify any degradation. No further water transfer events occurred following placement of the SFP gate.

- November 2, 2007, Units 1, 2, and 3 notice of unusual event declaration for detection of a credible threat when an explosive device was detected during a vehicle search at the site access check point as described in Event Notification 43764. This event was retracted based upon investigations performed by law enforcement personnel and Palo Verde Security Department, which determined that there was no explicit nor credible threatened action as described in 10 CFR 73.71, Appendix G, Paragraph I.

- November 22-25, 2007, Unit 1, failure of BOP ESFAS load sequencer Train A, that led to a unit shutdown as described in PVARs 3099500 and 3099679

- On October 6, 2007, Unit 2 manual trip from 100 percent power due to high SG sodium concentrations. The control room supervisor diagnosed an uncomplicated reactor trip. The SM and shift technical advisor reviewed Procedure EPIP-99 and determined no event classification was required. The sodium ingress was due to a leak in the condenser air removal Pump D seal water cooler.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

LER Report Reviews

a. Inspection Scope

The inspectors reviewed the below listed LER and related documents to assess: (1) the accuracy of the LER, (2) the appropriateness of corrective actions, (3) violations of requirements, and (4) generic issues.

(Closed) LER 05000528/2006004-00, “Technical Specification Required Shutdown on Failure of Class Pressurizer Heaters to be Able to Meet Their Mission Time”

Following the twelfth Unit 1 refueling outage, in which the pressurizer heaters were replaced, the licensee observed several failures of individual heaters due to electrical shorts or degradations in electrical insulation resistance and began an enhanced monitoring program for these heaters. On September 18, 2006, during the increased monitoring of the Unit 1 pressurizer heaters, the megger results were low on the Class 1E pressurizer heaters. The licensee determined that the Class 1E pressurizer heaters may not meet their mission time, declared both banks of pressurizer heaters inoperable, and shutdown the reactor to replace the heater elements. The vendor evaluation concluded that the heaters failed from stress corrosion cracking due to a manufacturing defect. An additional concern involved some heaters operating at a voltage higher than originally specified and that this may have caused the heater sheath to exceed its design temperature limit of 800°F. The vendor issued Nuclear Services Advisory Letter 07-8 on exceeding the design temperature limit. The inspectors reviewed this LER and no findings of significance were identified and no violation of NRC requirements occurred. The licensee documented the failed pressurizer heaters in CRDR 2914478. This LER is closed.

b. Findings

No findings of significance were identified.

Personnel Performance

a. Inspection Scope

The inspectors: (1) reviewed operator logs, plant computer data, and/or strip charts for the below listed evolutions to evaluate operator performance in coping with nonroutine events and transients, (2) verified that operator actions were in accordance with the response required by plant procedures and training, and (3) verified that the licensee has identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the nonroutine evolutions sampled.

- On October 30, 2007, Unit 2, a loss of letdown occurred due to a high letdown heat exchanger outlet temperature. This high temperature resulted in the closure of containment letdown isolation Valve CHB-UV-0523. Operations personnel appropriately implemented abnormal operating Procedure 40AO-9ZZ05, “Loss of Letdown,” Revision 18, to restore pressurizer level to the required band and recover the letdown system. The chemical volume and control system operated within design specifications following restoration of the letdown subsystem. Troubleshooting efforts were unable to determine the cause of the high letdown heat exchanger outlet temperature and subsequent loss of letdown event. This event was documented in CRDR 3086159.

- On December 3, 2007, the Unit 3 RCS level increased due to an unintentional transfer of 8000 gallons from the refueling water tank while restoring containment spray Train B. Control room personnel were also performing a shutdown cooling system vortex test on low pressure SI Train A. Operations personnel ensured that the suction paths were closed on containment spray Train B, but did not recognize that a gravity flowpath existed from the refueling water tank into the RCS. Investigation of ongoing evolutions by control room personnel identified the gravity flowpath and closed refueling water tank to Train B SI pumps suction Valve 3-CHB-530 to stop the water transfer and stabilize RCS level. There was no fuel in the reactor at the time of the event. This event was documented in CRDR 3105078.

- On November 22, 2007, Unit 1 experienced a loss of the BOP ESFAS load sequencer Train A. Erroneous indications at the BOP ESFAS cabinet included the EDG Train A running with the output breaker closed and the exhaust fan running. Operations personnel verified indications at the control panel, and locally, to determine that EDG Train A was not running and that the output breaker was open. The EDG Train A room exhaust fan was not running, but should have been for the observed sequencer status. The exhaust fan was subsequently determined to have tripped due to an invalid load shed signal from the malfunctioning sequencer. The malfunctioning load sequencer also caused the essential spray pond Pump A to start. Operations personnel appropriately identified affected equipment, diagnosed the event, and entered the applicable TS. Troubleshooting efforts determined the cause of the failure to be a failed relay and suppression diode in the circuitry. This event was documented in PVAR 3099500.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

Temporary Instruction 2515/166, "Pressurized Water Reactor Containment Sump Blockage," Palo Verde Units 1 and 3 (Closed)

Temporary Instruction 2515/166 was performed at Palo Verde Nuclear Generating Station, Unit 1, during May 2007, and documented in NRC Inspection Report 05000528; 05000529; and 05000530/2007003. Subsequent inspection of Unit 1, along with Unit 3, is documented in this report. The inspection phase of Temporary Instruction 2515/166 for Units 1 and 3 are complete. Temporary Instruction 2515/166 will be performed on Palo Verde Unit 2 during the Spring 2008 refueling outage.

The inspectors observed the physical installation of the sump strainers as committed to in the licensee's response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors." No concerns with the physical modifications were identified. In addition, the inspectors completed a walkdown of the containment area tags, labels, and coatings. The inspectors also reviewed the licensee's procedures and programs for accounting for and controlling equipment tags, latent debris, unqualified coatings, and chemicals inside containment. Programs to identify the scope of equipment tags, coatings, debris, and chemicals that have the potential to cause screen blockage were adequate, and the licensee has made needed changes to relevant procedures to control introduction of these items in the future.

At the time of the inspection, head loss and downstream effects testing were complete; however, the final evaluation reports for head loss and downstream effects were not available to the inspectors. The evaluation reports were not available because of a number of issues with vendor resources and revisions of the methodologies for performing the tests. Because of the delays, Palo Verde was granted an extended expiration date for the final response on December 27, 2007. The final response to Generic Letter 2004-002 is due June 30, 2008.

Listed below are the commitments and actions taken by Palo Verde Units 1 and 3.

.1 Evaluate the recommendations contained in the Westinghouse downstream effects evaluation for Palo Verde and establish an implementation schedule for appropriate recommendations.

Actions Taken

This commitment was completed on December 31, 2005. The licensee reviewed

Westinghouse WCAP-16406-P, June 2005, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191.” Any deviations from this evaluation were documented in Attachment 1 of the licensee’s September 1, 2005, response to Generic Letter 2004-02.

- Perform confirmatory head loss testing of new strainer with plant specific debris to ensure an adequate design.

Actions Taken

Head loss testing, performed by Control Components, Inc. and Sargent and Lundy, were completed in March 2007. Currently, the test reports are in review. The testing found that head loss of the strainer is greater than previously evaluated. Although head loss is greater, the preliminary results and further calculations indicate there is a net positive suction head margin of approximately 4 feet at elevated loss of coolant accident temperatures. Final resolution of the increased head loss is expected to be completed before June 30, 2008.

- Verify that a capture ratio of 97 percent or higher can be achieved in the final design of the new sump screen to ensure that the fuel evaluation contained in the Westinghouse downstream effects evaluation is bounding.

Actions Taken

In the licensee’s GL 2004-02 response, it states that a capture ratio of 97 percent must be achieved in order to prevent the creation of a thin bed on the underside of the fuel bottom nozzle following a hot leg break loss of coolant accident. A 97 percent capture ratio would ensure that the fuel evaluation in the Westinghouse downstream effects evaluation is bounding. During the March 2007 chemical and downstream effects testing, results of the tests indicate the strainers would achieve a capture ratio of 90 to 95 percent, which is not bounded by the Westinghouse evaluation. The licensee stated that the capture ratio for the sump strainer is lower than expected because there is a low amount of fiber in containment. In addition, testing has confirmed the capture ratio of the strainers will gradually increase when captured debris performs the “capturing” role through the accident duration. The licensee has contracted with Westinghouse for final resolution of the capture ratio issue of the sump strainers. This evaluation is expected to be complete before June 30, 2008.

- Perform sump strainer structural evaluation to ensure seismic and operational integrity.

Actions Taken

The structural evaluation for the new sump strainers was completed on October 31, 2006. This evaluation is applicable to Units 1, 2, and 3.

- Validate allocated margins for chemical effects in strainer head loss to ensure an adequate design.

Actions Taken

Validation of the allocated margin for chemical effects was tested in March 2007. Currently, the results of the test are being reviewed for validity. The review will be completed before June 30, 2008.

.2 Perform a confirmatory containment latent debris walkdown of Units 1 and 3.

Actions Taken

Latent debris walkdowns for Units 1 and 3 were completed by the licensee on June 30, 2006. A walkdown of Units 1 and 3 was completed by the inspectors in May 2007 and December 2007, respectively. The debris and head loss evaluations conservatively use 200 pounds per unit for transportable debris. A walkdown in Unit 2 identified 119 pounds of latent debris using NEI 04-07 sampling methods. Subsequent walkdowns were performed in Units 1 and 3 and identified that latent debris was within the bounds of the evaluations.

On October 26, 2007, as part of WO 3034098, the licensee conducted a containment walkdown of Unit 1 to quantify and remove susceptible tape and flex conduit. From the walkdown, the licensee preliminarily estimated that 600-square feet of combined tape and conduit, or transient debris, had not been accounted for in the sump loading analysis. The licensee initiated PVAR 3083224 to evaluated the condition. Subsequent evaluation (OD 2508658) determined that the total transient debris was 298-square feet.

During testing of the sump strainers, as documented in Control Components, Inc. Test Report title, *Chemical Effect Head Loss Specification Palo Verde NPP*, the sump strainer area was reduced by 400-square feet to accommodate miscellaneous debris. Since the total transient debris was bounded by the reduced area during testing, the licensee determined the sump strainers were operable.

.3 Perform a confirmatory containment unqualified coating walkdown of Units 1 and 3.

Actions Taken

The containment coating walkdown was completed by the licensee before June 30, 2006. A walkdown of Units 1 and 3 was completed by the inspectors in May 2007 and December 2007, respectively. All unqualified coatings are maintained in an “unqualified coatings log” per the licensee’s procedure. The licensee’s debris generation calculation assumes that all coatings in the zone-of-influence are transported to the sump as fine debris.

.4 Review the existing programmatic controls for containment coatings identified in the response to GL98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," for their adequacy.

Actions Taken

The licensee completed the review of programmatic controls for containment coatings and enhanced the procedure before November 30, 2006. The coatings procedure is applicable to all three units. The inspectors reviewed the procedures. There were no issues identified.

.5 Review the existing programmatic and procedural controls in place to prevent potentially transportable debris in the containment building to ensure that the bounding assumptions in the design of the new strainers will be maintained.

Actions Taken

The licensee completed the review of programmatic controls for containment coatings and enhanced the procedure before November 30, 2006. The coatings procedure is applicable to all three units. The inspectors reviewed the procedures. There were no issues identified.

.6 Implement in Unit 1 changes to programs and procedures to ensure and/or enhance the control of transportable debris in containment.

Actions Taken

The licensee has completed changes to programs and procedures to ensure and enhance the control of transportable debris in containment. The inspectors reviewed the procedures. There were no issues identified. The licensee has added restrictions to their procedures for tags, insulation, and additional debris. If an item has been qualified for continuous use in containment, the licensee's procedure requires the item to be included in the debris loading evaluation. In addition, the description and location of potential debris will be posted outside of containment.

.7 Implement in Unit 2 changes to programs and procedures to ensure and/or enhance the control of transportable debris in containment.

Actions Taken

The licensee has completed changes to programs and procedures to ensure and enhance the control of transportable debris in containment. The licensee has added restrictions to their procedures for tags, insulation, and additional debris. If an item has been qualified for continuous use in containment, the licensee's procedure requires the item to be included in the debris loading evaluation. In addition, the description and location of potential debris will be posted outside of containment.

.8 Implement in Unit 3 changes to programs and procedures to ensure and/or enhance the control of transportable debris in containment.

Actions Taken

The licensee has completed changes to programs and procedures to ensure and enhance the control of transportable debris in containment. The licensee has added restrictions to their procedures for tags, insulation, and additional debris. If an item has been qualified for continuous use in containment, the licensee's procedure requires the item to be included in the debris loading evaluation. In addition, the description and location of potential debris will be posted outside of containment.

.9 Install larger sump strainers in Unit 1.

Actions Taken

Larger sump strainers were installed in Palo Verde Unit 1 during the May/June 2007 refueling outage.

.10 Install larger sump strainers in Unit 2.

Actions Taken

Palo Verde Unit 2 was granted an extension to implement the sump modifications after the December 31, 2007, due date. New sump strainers will be installed during the April/May 2008 refueling outage. Inspectors will complete Temporary Instruction 166 for Unit 2 at a later date.

.11 Install larger sump strainers in Unit 3.

Actions Taken

Larger sump strainers were installed in Palo Verde Unit 3 during the October/November 2007 refueling outage.

.12 Remove installed Fiberfrax insulation in Units 1, 2, and 3.

Actions Taken

All Fiberfrax insulation in Palo Verde Units 1, 2, and 3 has been removed.

.13 Remove installed Fiberfrax insulation in Unit 2.

Actions Taken

All Fiberfrax insulation in Palo Verde Units 1, 2, and 3 has been removed.

.14 Remove installed Fiberfrax insulation in Unit 3.

Actions Taken

All Fiberfrax insulation in Palo Verde Units 1, 2, and 3 has been removed.

.15 After plant specific strainer testing has been completed and the Westinghouse downstream effects evaluation for Palo Verde has been evaluated, APS will submit an update to the NRC to report the validation of the allocated margins for chemical effects and identify any recommendations from the Westinghouse evaluation to be implemented.

Actions Taken

This report will be submitted no later than June 30, 2008.

4OA6 Meetings, Including Exit

On October 26, 2007, the inspectors presented the occupational radiation safety inspection results to Mr. M. Perito, Plant Manager, Nuclear Operations, and other members of the licensee's management staff who acknowledged the findings. On November 30, 2007, the inspectors presented additional occupational radiation safety inspection results to Mr. R. Bement, Vice President, Nuclear Operations, and other members of the licensee's management staff who acknowledged the findings.

On November 29, 2007, the inspectors presented the inspection results of the licensed operator annual requalification examination with Mr. W. Potter, Training Supervisor. A telephone exit meeting was held with Mr. W. Potter, on November 29, 2007. The licensee acknowledged the findings presented in both the briefing and the final exit meeting.

On January 9, 2008, the inspectors presented the resident inspection results to Mr. R. Edington, Executive Vice President, Nuclear, and other members of the licensee's management staff who acknowledged the findings presented.

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

4OA7 Licensee-Identified Violations

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

- Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 9.a, requires, in part, procedures for performing maintenance that can affect the performance of safety-related equipment. Contrary to this requirement, on October 6, 2007, as part of the work for relocating Valve SIA-UV-651 during Refueling Outage 3R13, the operability of SDC Train A was adversely affected. Bechtel workers, using design implementing WO 2914425, removed Hanger 13-RC-071-H-00E in accordance with the instructions. Design implementing WO 2914425 was authorized to work in Mode 5, 6, or defueled. At the same time, Hanger 3-RC-071-H-00F was inadvertently removed since Hangers 13-RC-071-H-00E and 3-RC-071-H-00F had a common piece of structural steel. Design implementing WO 2914425 specified that Hangar 3-RC-071-H-00F was to be removed only when the reactor was defueled. This condition was discovered by an engineer on October 6, 2007, when there was fuel in the reactor and SDC Train A was in service. Nonconformance Report 25030-U3-040 was generated when the condition was identified. Approximately 14 hours later, PVAR 3072732 was generated on this issue, and went to the control room for review. Following review of the PVAR, SDC Train A was declared inoperable. An evaluation determined that SDC Train A was operable with Hanger 3-RC-071-H-00F removed. CRDR 3080110 was written for the lack of timeliness between the identification of the adverse condition and the notification to the Unit 3 control room. Findings associated with the untimely notification of this condition to the affected control room are documented in Inspection Report 05000528; 05000529; 05000530/2007012. This finding is determined to have very low safety significance because the finding does not result in noncompliance with low temperature over pressure protection TSs, nor does it degrade the ability of containment to remain intact following an accident. Additionally, the finding does not degrade the licensee's ability to terminate a leak path, add RCS inventory, recover decay heat removal once it is lost, or establish an alternate core cooling path. Lastly, the finding does not increase the likelihood of a loss of RCS inventory, decay heat removal, or offsite power.

- Technical Specification 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 9.a, requires, in part, procedures for performing maintenance that can affect the performance of safety-related equipment. Procedure 40DP-9ZZ17, "Control of Doors, Hatches, and Floor Plugs," Revision 40, Appendix A, Step 2, states that "[In Table 8] a "Yes" appearing in a column identifies the barrier as a controlled barrier. All identified Responsible Organization(s) shall be contacted for compensatory measures and authorization prior to blocking open or removing the barrier." Contrary to this requirement, on December 3, 2007, during shiftly rounds, security personnel found a hose routed through Door Y-1-06 for maintenance, with no compensatory measures established.
  Procedure 40DP-9ZZ17, Appendix A, Table 8, identifies Door Y-1-06 as a controlled barrier requiring operations personnel to be contacted for compensatory measures and authorization prior to blocking open this door. Both trains of engineered safety feature PREACS were declared inoperable in accordance with TS 3.7.13. This condition was immediately corrected and PREACS was restored to operable status on December 3, 2007. The licensee entered this item into the CAP as PVAR 3103619. After further evaluation and testing by engineering personnel, routing a hose through Door Y-1-06 would not have rendered PREACS inoperable, and the system would have been able to perform its design function to maintain the auxiliary building envelope, below the 100 foot elevation, under a measurable negative pressure. This finding is determined to have very low safety significance because it does not represent a loss of system safety function and the finding does not screen as risk significant due to a seismic, flooding, or severe weather initiating event. The failure to follow Procedure 40DP-9ZZ17 to provide compensatory measures for blocked open doors is a reoccurring event documented in LER 05000530/2006001-00, "Two Independent Trains of Auxiliary Feedwater Inoperable," LER 05000529/2006002-00, "Two Independent Trains of Auxiliary Feedwater Inoperable Due to a Single Cause," and as a licensee-identified finding in NRC Inspection Report 05000528; 05000529; and 05000530/2006004.

- Technical Specification 5.4.1.a requires procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends, in Section 7, procedures for access control to radiation areas including a radiation work permit system. Implementing Procedure 75DP-9RP01, “Radiation Exposure and Access Control,” Revision 10, Section 2.1.6, required that workers read the applicable radiation exposure permit for their specific job or task, and obey all instructions and requirements. Radiation Exposure Permit 3-6016A required individuals to receive a prejob brief by radiation protection and review current radiological survey data for the work area prior to entry during radiography operations. Contrary to the above, on October 30, 2007, the licensee identified that an individual had entered a posted radiography area boundary without receiving a prejob brief by radiation protection and reviewing current radiological survey data prior to entering the work area. The licensee documented the occurrence in CRDR 3086532. The violation is determined to have very low safety significance because it was not associated with ALARA planning or work controls issues, there was no overexposure or a substantial potential for an overexposure, and the ability to assess dose was not compromised.

- Technical Specification 5.4.1.a requires procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends, in Section 7, procedures for access control to radiation areas including a radiation work permit system. Implementing Procedure 75DP-9RP01, “Radiation Exposure and Access Control,” Revision 10, Section 2.1.6, required that workers read the applicable radiation exposure permit for their specific job or task, and obey all instructions and requirements. Radiation Exposure Permit 3-3415A, “Perform Pressurizer and RCS Nozzle Weld Overlays,” Task 2 required, in part, that workers have operating electronic dosimeters while working on the lower pressurizer. Contrary to the above, on November 16, 2007, a radiation protection technician found that a worker’s electronic dosimeter was not turned on and therefore not operating as required by radiation exposure permit. The licensee documented the occurrence in CRDR 3095065. The violation is determined to have very low safety significance because it was not associated with ALARA planning or work controls issues, there was no overexposure or a substantial potential for an overexposure, and the ability to assess dose was not compromised.

- Technical Specification 5.4.1.a requires procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends, in Section 7, procedures for access control to radiation areas including a radiation work permit system. Implementing Procedure 75DP-9RP01, “Radiation Exposure and Access Control,” Revision 10, Section 2.1.6, required that workers read the applicable radiation exposure permit for their specific job or task, and obey all instructions and requirements. Radiation Exposure Permit 3-35021, “Valve, Flange, and Pump Maintenance and Inspection,” Task 2 states, in part, that contaminated system breaches require continuous radiation protection coverage. Contrary to the above, on November 19, 2007, a radiation protection technician found that workers had cut into and breached a contaminated system to remove a valve without notifying radiation protection for continuous coverage as required by Radiation Exposure Permit 3-3502I. The licensee documented the occurrence in CRDR 3101678. The violation is determined to have very low safety significance because it was not associated with ALARA planning or work controls issues, there was no overexposure or a substantial potential for an overexposure, and the ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Andrews, Director, Performance Improvement
S. Bauer, Department Leader, Regulatory Affairs
J. Bayless, Senior Engineer
R. Bement, Vice President, Nuclear Operations
P. Borchert, Director, Operations
P. Brandjes, Department Leader, Maintenance
R. Buzard, Senior Consultant, Regulatory Affairs
D. Carnes, Director, Nuclear Assurance
P. Carpenter, Department Leader, Operations
R. Cavalieri, Director, Outages
K. Chavet, Senior Consultant, Regulatory Affairs
D. Coxon, Unit Department Leader, Operations
R. Eddington, Executive Vice President, Nuclear
D. Elkington, Consultant, Regulatory Affairs
T. Engbring, Senior Engineer
J. Gaffney, Director, Radiation Protection
T. Gray, Department Leader, Radiation Protection
K. Graham, Department Leader, Fuel Services
M. Grigsby, Unit Department Leader, Operations
D. Hautala, Regulatory Affairs
D. Hansen, Senior Consulting Engineer
R. Henry, Site Rep., Salt River Project
J. Hesser, Vice President, Engineering
R. Indap, Senior Engineer
M. Karbasian, Director, Engineering
W. Lehman, Senior Engineer
S. McKinney, Department Leader, Operations Support
E. O'Neil, Department leader, Emergency Preparedness
M. Perito, Plant Manager, Nuclear Operations
F. Poteet, Senior Inservice Inspection Engineer
M. Radspinner, Section Leader, Systems Engineering
T. Radtke, General Manager, Emergency Services and Support
H. Ridenour, Director, Maintenance
F. Riedel, Director, Nuclear Training Department
J. Scott, Section Leader, Nuclear Assurance
M. Shea, Director, Safety Culture/Impact
E. Shouse, Representative, El Paso Electric
M. Sontag, Department Leader, Performance Improvement
K. Sweeney, Department Leader, Systems Engineering
J. Taylor, Nuclear Project Manager, Public Service of New Mexico
J. Taylor, Unit Department Leader, Operations

D Vogt, Section Leader, Shift Technical Advisors, Operations
T. Weber, Section Leader, Regulatory Affairs
J. Wood, Department Leader, Nuclear Training Department

NRC Personnel
M. Runyan, Senior Reactor Analyst, Region IV

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000528/2007005-01	NCV	Failure to Take Adequate Corrective Actions to Prevent Recurrence of Significant Condition Adverse to Quality (Section 1R12)
05000528; 05000529/2007005-02	NCV	Two Examples of a Failure to Properly Implement the Operability Determination Process (Section 1R15)
05000528/2007005-03	NCV	Two Examples of a Failure to Post and Control a High Radiation Area (Section 2OS1)
05000528/2007005-04	NCV	Failure to Evaluate the Radiological Hazard Caused by Decontamination (Section 2OS1)
05000530/2007005-05	NCV	Failure to Follow Procedural Guidance and Radiation Work Instructions (Section 2OS1)
05000528; 05000529; 05000530/2007005-06	NCV	Two Examples of Inadequate Design Controls for Refueling Machine (Section 4OA2)

Closed

05000528/2006004-00	LER	Technical Specification Required Shutdown on Failure of Class Pressurizer Heaters to be able to meet their Mission Time (Section 4OA3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

Section 1R01: Adverse Weather Protection

Procedures

NUMBER	TITLE	REVISION
40OP-9ZZ17	Cold Weather Protection	32
41OP-1OP01	Manual Operation of Air Operated Valves Appendix H	29
42OP-2OP01	Manual Operation of Air Operated Valves Appendix H	26

PVARs

2947058 2953367 3089256

Miscellaneous

UFSAR

Section 1R04: Equipment Alignment

Procedures

NUMBER	TITLE	REVISION
40ST-9AF07	Auxiliary Feedwater Pump AFA-P01 Monthly Valve Alignment	4
40ST-9AF08	Auxiliary Feedwater Pump AFB-P01 Monthly Valve Alignment	3
40OP-9AF01	Appendix A, Essential Auxiliary Feedwater System Electrical Alignment List	39
40OP-9AF01	Appendix B, Essential Auxiliary Feedwater System Valve Alignment List	39
40OP-9AF01	Appendix C, Essential Auxiliary Feedwater System Control Board Alignment List	39
40OP-9AF01	Appendix D, Essential Auxiliary Feedwater System Instrument Alignment List	39

Drawings

NUMBER	TITLE	REVISION
13-M-AFP-001	P&I Diagram, Auxiliary Feedwater System	34
13-M-CTP-001	P&I Diagram, Condensate Storage and Transfer	20
01-M-SGP-001	P&I Diagram, Main Steam System	56

03-M-SIP-001	P&I Diagram, Safety Injection & Shutdown Cooling System	31
03-M-PCP-001	P&I Diagram, Fuel Pool Cooling and Cleanup System	23
03-M-NCP-002	P&I Diagram, Nuclear Cooling Water System	8
03-M-NCP-001	P&I Diagram, Nuclear Cooling Water System	7

Miscellaneous

UFSAR, Section 10.4.9, Auxiliary Feedwater System, Revision 14
Design Basis Manual, Revision 16
System Health Report, January 1, 2007, through June 30, 2007

Section 1R05: Fire Protection

Procedures

NUMBER	TITLE	REVISION
30DP-9WP11	Scaffolding Instructions	18
30DP-9WP11	Scaffolding Instructions	5
30DP-9WP11	Scaffolding Instructions	6
14DP-0FP31	Fire System Impairment	12
14DP-0FP33	Control of Transient Combustibles	15
14FT-9FP42	Monthly Portable Fire Extinguisher Inspection	9
14FT-9FP13	Fire Hose Station Operational and hydrostatic Test	8
14FT-9FP12	Fire Hose Station Inspection,	9

Drawings

NUMBER	TITLE	REVISION
13-VTD-P115-0008	Peerless Miscellaneous Drawings and Parts List - Diesel Driven Fire Pump	2

CRDRs

3075291

PVARs

2978177 3010313 3072242 3089381 3105292

Miscellaneous

Pre-Fire Strategies Manual, Revision 18

Fire System Component Condition Records 3038447 and 3045163

Technical Requirements Manual 3.11, Revision 39

Specification 13-CN-0380, Installation Specification For Seismic Category IX and Non-Seismic Scaffolding, Revision 9

Arizona Public Services History Report, Unit 1 Extinguishers

Engineering Evaluation Request 91-FP-011, dated December 24, 1991

National Fire Protection Association 1962, Standard for the Inspection, Care, and Use of Fire Hose, Couplings, and Nozzles and the Service Testing of Fire Hoses, 2003 Edition

PVNGS Pre-Fire Strategies Manual, Revision 19

UFSAR Appendix 9B, Fire Protection Evaluation Report

UFSAR Section 9.5.1, Fire Protection Evaluation Report

Section 1R06: Flood Protection Measures

Procedures

NUMBER	TITLE	REVISION
14FT-9FP70	Appendix R and Former Technical Specification Penetration Seal Surveillance	7
31MT-9ZZ12	Replacement/Rework of Penetration and Internal Conduit Seals	7
81DP-0ZZ01	Civil System, Structure, and Component Monitoring Program	12

Drawings

NUMBER	TITLE	REVISION
01-M-OWP-003	P&I Diagram, Oily Waste and Non-Radioactive Waste System (Control Building)	6

CRDRs

2846647 2882166 2970134

PVARs

2968359

WOs

3056337 3056342 3056345

CRAIs

2970135 3007692 3007697 3007702 3007708 3007710 3045868

Miscellaneous

Calculation 13-MC-ZJ-200, “Control Building Flooding,” Revision 6

NRC Information Notice 2005-30, “Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design,” dated November 7, 2005

NRC Information Notice 2007-01, “Recent Operating Experience Concerning Hydrostatic Barriers,” dated January 31, 2007

UFSAR, Section 3.4, Water Level (Flood) Design

UFSAR, Section 9.3.3, Equipment and Floor Drainage Systems

Procedures

NUMBER	TITLE	REVISION
WPS 03-08-T-804 Bottom	Welding Procedure Specification, Pressurizer Weld Overlay, Automated Gas Tungsten Arch Welding	0
WPS P1(G2)-T(RA)	Bechtel Welding Procedure Specification, RCS	5
WPS P1(G2)-T(RA)	Bechtel Welding Procedure Specification, RCS Clad	0
70TI-9ZC01	Boric Acid Walkdown Leak Detection	6
RT-ASME/ANSI	ASME Sec III Class II Welds	3
73TI-9ZZ78	Visual Examination for Leakage	8
73DP-9XI04	Inservice Inspection Program System Pressure Testing Administrative Requirements	3
73TI-9ZZ07	Penetrant Examination	12

CRDRs

2884926	2900046	2920113	2966039	3010261
2890601	2905368	2934501	3005292	
2894556				

Drawings

Welding Diagram, “Auxiliary Feedwater Steam Generator #1,” Revision 0

Section 1R11: Licensed Operator Requalification Program

CRDRs

3079211 3079295 3105121

CRAIs

3079211 3079212 3079299 3104144 3105123

PVARs

3077329 3077920 3104069

Miscellaneous

SES-0-08-F-00, LOP/LOOP/Blackout, Licensed Operator Continuing Training Simulator Evaluation Guide

Emergency Plan Implementing Procedure EPIP-99, Appendices A and P, Revision 16

Simulator Evaluation Summary Sheet, Crew 14, Cycle NLR0705

SES-0-03-T-00, Inadvertent AFAS/ATWS/LOCA, Licensed Operator Continuing Training Simulator Evaluation Guide

Simulator Evaluation Summary Sheet, Crew 34, Cycle NLR0705

Licensed Operator Continuing Training Program Description, Revision 34

Licensed Operator Continuing Training 2007 - 2008 Two Year Schedule, Revision 4

Remedial Training Notification and Action Plans

Section 1R12: Maintenance Effectiveness

Procedures

NUMBER	TITLE	REVISION
70DP-0MR01	Maintenance Rule	15
70DP-0MR01	Maintenance Rule	17
01DP-9ZZ01	Systematic Troubleshooting	0
40OP-9AF01	Essential Auxiliary Feedwater System	37
73ST-9AF02	AFA-P01 Inservice Test	39
ODP-1	Operations Department Practices, Control Room Monitoring	06

Drawings

NUMBER	TITLE	REVISION
01-P-AFF-131	MSSS Isometric Turbine Driven Pump Auxiliary Feedwater System, Sheets 1 - 4	
1031210-2	Target Rock, Solenoid Operated Valve High Temp & Press Energize to Open (FC) Sizes 3/8" Thru 2", Sheets 1 & 2	E
01-M-SGP-001	P & I Diagram Main Steam System, Sheets 1 & 2	56

CRDRs

2835245 2883283 3020226 3064675 3078032

PVARs

3052708 3064151 3070147 3076744

CRAIs

3065717 3066009 3073854 3078033 3080275 3080457 3083320

WOs

2950563	3077684	3081037	3083917	3083919
3064157	3080749	3081153	3083918	
3076750	3080838	3083900		

Miscellaneous

January 1 - June 30, 2007, QSPDS System Health

Preliminary Investigation Information Regarding the Failure of Unit 1, SG 2 Steam Admission Valve to AFW Pump Turbine Valve SGA-UV-138A, dated September 21, 2007

Kepner-Tregoe Problem Analysis Detailed Report, dated October 26, 2007

Significant Root Cause Investigation Report, CRDRs 3064675 and 3078032, Valve SGA-UV-138A Repeat Failure, Event Dates: dated September 17 and October 15, 2007,

Tagging Permits 145596, 145642, 145643

Engineering Game Plan, Level B, Valve SGA-UV-138A, as documented in PVAR 3076744 and CMWO 3076750, Revisions 0 through 5

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

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51DP-9OM03	Site Scheduling	18
40DP-9OP02	Conduct of Shift Operations	37
70DP-9MR01	Maintenance Rule	17
ODP-15	Operations Department Practices	1
73ST-9AF02	AFA-P01 Recirculation Flow - Inservice Test	40
40OP-9DG02	EDG B	49
40ST-9DG02	Diesel Generator B Test	34
40DP-9OP06	Operations Department Repetitive Task Program Appendix SA001	96
73ST-9AF02	AFA-P01 Recirculation Flow - Inservice Test	40

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01-M-DFP-001	P & I Diagram - Diesel Fuel Oil and Transfer System	11
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ELE 342-0107	Sys Intcon Diag-Load Sequencer/Tester Train A, Sheet 1	7
ELD 342-1510	Schematic-Relay Module, Sheet 1	1
ELD 342-1500	Module Assy-Relay, Sheet 1	1
ELE 342-5410	Schematic - ESF Load Sequencer / Auto Test Module Sheets 1, 2, and 3	7

<u>CRDRs</u>

2928626 2950136 3064675 3078032

<u>WOs</u>

2969643	3064157	3081153	3099502
2995731	3076750	3092613	

<u>CRAIs</u>

3066009 3094567

<u>PVARS</u>

2948762	3070147	3092611	3099500
3064151	3076744		

<u>Miscellaneous</u>

Scheduler's Evaluation for Unit 1, November 13 and 14, and December 4, 2007

Technical Specification Component Condition Records 3104228, 3104232, and 3104234

Permits 139330, 139332, 139333, 140135, 141063, 145120, 145121, 145155, 145416, 147541

Technical Specification 3.7.5, Auxiliary Feedwater System

White Paper, Preliminary Investigation Information Regarding the Failure of the Unit 1 SG 2 Steam Admission Valve to AFW Pump Turbine Valve SGA-UV-138A, dated September 21, 2007

Prompt Operability Determination, Malfunction of Unit 1 A SG 2 Steam Bypass Valve to AFW Pump Turbine Valve SGA-UV-138A, Revision 0

Technical Specification 3.8.1, AC Sources - Operating

Technical Specification 3.8.3, Diesel Fuel Oil, Lube Oil and Starting Air

UFSAR, Section 8.3.1, AC Power Systems

Alarm Typer Printout, November 22, 2007, Time 19:18-19:19

Engineering Root Cause of Failure Analysis, Level 1, EDG B Flange Leak on 5-L Fuel Injection Pump, February 20, 2007

Failure Analysis of Fuel Injection Pump, Revision 0, dated February 5, 2007

White Paper, EDG Fluid Leakage and Operability, Revision 1

Engine Combustion Report APS Emergency Diesel Generator, November 14, 2007

Unit 1 Shift Logs, November 13-14 and 22-25, 2007

Troubleshooting Plan For BOP ESFAS Train A Failure in Unit 1, Revisions 0-4

White Paper, Failure Progressive Mechanism For Diode

Section 1R15: Operability Evaluations

Procedures

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40DP-9OP26	Operability Determination and Functional Assessment,	18
40DP-9OP08	Diesel Generator Test Record	44

Drawings

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CRDR

3110947

DFWO

3084550

WOs

3079030 3085714 3109610 3111026

PVARs

3078055 308443 3109083 3109481 3109607

Miscellaneous

Palo Verde Design Input Requirements Checklist
Palo Verde Independent Verification Checklist
Palo Verde Impact Review Form
Palo Verde Environmental Screening Form
S-07-0379, 10 CFR 50.59 Screening Form, Revision 0
Unit 1 Operating Logs, October 28-30, 2007
Operations Department Practices

Section 1R17: Permanent Plant Modifications

Procedures

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13-Q-ZZP-001	EQ Configuration Drawing Limitorque Valve Actuator	0
13-E-ZCC-007	Containment Bldg Conduit & Tray Plan at El. 80 ft Level A ZCAA, ZCAB	27
03-M-SIP-002	P & I Diagram Safety Injection and Shutdown Cooling System	19

PVARs

3098350 3102047 3103628 3103823 3104219 3105747

3050727

WOs

2976219 3054112

Miscellaneous

S-07-0329, “10 CFR 50.59 Screening and Evaluation for DMWO 3054112,”Revision 0
Unit 3 Reactor Vessel Level Monitoring System flow induced level error data
E-07-011, “10 CFR 50.59 Screening and Evaluation,” Revision 0
Design Modification 2541284, “Steam Generator Change Out”

Section 1R19: Postmaintenance Testing

Procedures

NUMBER	TITLE	REVISION
40DP-9OP06	Operations Department Repetitive Task Program Appendix SA001	96
40OP-9DG02	EDG B	49
40ST-9DG02	Diesel Generator B Test	34
30DP-9WP04	Post-Maintenance Testing Development	14
30DP-9MP09	Preventive Maintenance Processes and Activities	20

Drawings

NUMBER	TITLE	REVISION
ELE 342-0107	System Intcon Diag-Load Sequencer/Tester Train A, Sheet 1	7
ELD 342-1510	Schematic-Relay Module, Sheet 1	1
ELD 342-1500	Module Assy-Relay, Sheet 1	
ELE 342-5410	Schematic-ESF Load Sequencer / Auto Test Module Sheets 1, 2, and 3	7
01-M-DFP-001	P & I Diagram - Diesel Fuel Oil and Transfer System	11
01-M-DGP-001	P & I Diagram - Diesel Generator System Sheet 1	48

PVARs

3099500 3099502

WOs

2914420 2939316 2995731

Miscellaneous

Troubleshooting Plan For BOP ESFAS Train A Failure in Unit 1, Revisions 0 - 4
White Paper, Failure Progressive Mechanism For Diode
Technical Specification 3.8.1, AC Sources - Operating
Failure Analysis of Fuel Injection Pump, Revision 0
White Paper, EDG Fluid Leakage and Operability, Revision 1
Engine Combustion Report, Emergency Diesel Generator, dated November 14, 2007
S-07-0360, 10 CFR 50.59 Screening and Evaluation, Revision 0

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Procedures

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70DP-0RA01	Shutdown Risk Assessments	22
40EP-9EO01	Standard Post Trip Actions	14
40EP-9EO02	Reactor Trip	8
40DP-9ZZ01	Containment Entry in Modes 1 Thru 4	26
72OP-9RX01	Calculation of Estimated Critical Condition, Appendices A, B, and C	20
40OP-9ZZ03	Reactor Startup	46
40OP-9ZZ01	Containment Entry in Modes 1 Thru 4	27

WO

3085473

PVARs

3084787 3086308

DFWOs

3085436

Permits

137458	139608	143273	143462	146500
139182	139609	143274	143494	
139567	142827	143405	145574	

Miscellaneous

Plant Transient Review Assessment, dated October 31, 2007

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Procedures

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73DP-9CL02	Containment Leakage Testing Program	20
73DP-9CL01	Containment Leakage Type “B” and “C” Testing	30

WO

3082177

Miscellaneous

PVNGS Surveillance Test Package Review Sheet
Surveillance Test Log for Procedure 73ST-9AF02

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Procedures

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81DP-0DC17	Temporary Modification Control	20

PVAR

3076979

Miscellaneous

Temporary Modification Request 03-91-CP-023

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Procedures

NUMBER	TITLE	REVISION
75DP-0RP01	Radiation Protection Program Overview	6
75DP-0RP02	Radiation Contamination Control	8
75DP-9RP01	Radiation Exposure and Access Control	10
75RP-0RP01	Radiological Posting and Labeling	24

75RP-9OP02	Control of High Radiation Areas, Locked High Radiation Areas and Very High Radiation Areas	21
75RP-9RP07	Radiological Surveys and Air Sampling	15
75RP-9RP10	Conduct of Radiation Protection Operations	21

Radiation Exposure Permits

1-8504F	3-3415A	3-3508J	3-6016A	9-1035A
3-1365	3-35021	3-6007	9-1009D	
3-2006F	3-3502G			

CRDRs

2957554	3046953	3077342	3081978	3098867
2965089	3056608	3077841	3086532	3101678
2970612	3070800	3081935	3095065	3101693
3007629	3072542	3081978	3098763	
3028825				

Miscellaneous

Self-Assessment Site Work Management System 2956957

Section 2OS2: ALARA Planning and Controls (71121.02)

CRDRs

2942482 3007629 3021261

Procedures

NUMBER	TITLE	REVISION
75DP-0RP01	Radiation Protection Program Overview	6
75DP-0RP02	Radioactive Contamination Control	8
75DP-0RP03	ALARA Program Overview	3
75RP-9RP02	Radiation Exposure Permits	18
75RP-9RP02	Radiation Contamination Control	8
73TI-0ZZ13	Radiographic Examination	14
75RP-9RP01	Radiation Exposure and Access Control	10

75RP-9RP10	Conduct of Radiation Protection Operations	24
73TI-0ZZ23	Digital Radiographic Examination	0

Section 4OA1: Performance Indicator Verification (71151)

CRDRs

3007629 3056608

Procedures

Number	TITLE	REVISION
75RP-0LC01	Performance Indicator Occupational Radiation Safety Cornerstone	2
75RP-0LC02	Performance Indicator Public Radiation Safety Cornerstone	2

Section 4OA2: Identification and Resolution of Problems (71152)

CRDRs

2852459 3030759 3068655 3068656

CRAIs

3030759	3048828	3049369	3068657	3083132
3048819	3048831	3049379	3068659	
3048823	3048833	3049429		

PVARs

3029619 3048775

WO

3049272

Drawings

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03-E-ZCB-004	Elementary Diagram, Containment Building System Refueling Service Requirements Cabling Block Diagram	6
03-E-NHA-028	Single Line Diagram 480 Volt Non-Class 1E Power System Motor Control Center 3E-NHN-M28	19

03-E-NHA-004	Single Line Diagram 480 Volt Non-Class 1E Power System Motor Control Center 3E-NHN-M04	6

Vendor Drawings

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C-21813-04-E	Refueling Machine Elementary Diagram, Sheet 2 of 4	9
C-21813-04-E	Refueling Machine Elementary Diagram, Sheet 3 of 4	9
C-21813-04-E	Refueling Machine Elementary Diagram, Sheet 4 of 4	9

Miscellaneous

Licensing Document Change Request 06-F027

Fuel Handling Design Basis Manual, Revision 6

S-06-0198, 10 CFR 50.59 screening and evaluation, Revision 1

N001-0503-00629, Refueling Machine Controls Upgrade Project Hoist Drive Assembly Drawing and Bill of Materials, Revision 1

N001-0503-00629, SFHM Controls Upgrade Project - Drive and Assembly, Revision 1

Refueling Machine Hoist Motor Namplate Data

Egineering Design Change 2007-00596

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion (71153)

Procedures

NUMBER	TITLE	REVISION
40AO-9ZZ23	Loss of Spent Fuel Pool Level or Cooling	10
40AO-9ZZ10	Condenser Tube Rupture	17
40EP-9EO01	Standard Post Reactor Trip Actions	14
40OP-9ZZ10	Mode 3 to Mode 5 Operations	54
74DP-9CY04	System Chemistry Specifications	51

CRDRs

2900393	2925633	2984700	3033543	3074272
2904740	2925806	2990092	3032677	3086158
2906158	2929277	3005058	3033623	
2914478	2974523			

PVARs

30777104 3101297 3101442

WOs

2304865 3032675 3085940 3099502

Miscellaneous

Unit 1 Control Room Logs, Units 1, 2, and 3

Chemical Volume Control System Temperature Trends

Emergency Notification 42847

Emergency Notification 43764

Westinghouse Proprietary Class 2 Root Cause Analysis Report, CAPs-RCA-06-265.M022, Revision 0

NSAL Letter 07-8, High Power Density Pressurizer Heater Sheath Design Temperature

Section 4OA5: Other Activities

Procedures

NUMBER	TITLE	REVISION
40ST-9ZZ09	Containment Cleanliness Program	17
40DP-9OP29	Power Block Permit and Tagging	32

Section 4OA7: Licensee-Identified Violations

Procedures

NUMBER	TITLE	REVISION
18FT-9FP31	Functional Test of Appendix A Fire Doors - Control Building 74', 100', 120', 140', and 160'	8
40DP-9ZZ17	Control of Doors, Hatches, and Floor Plugs	40

PVARs

3072229 3099448 3101869 3103619

CRDRs

107242	2860836	3075289	3104879
2837139	2918716	3104787	

CRAIs

3055290 3075290 3087995 3088004 3104879 3104882

WOs

2922324 2926239 2927578

Miscellaneous

Technical Specification 3.7.13, Engineered Safety Feature Pump Room Exhaust Air Cleanup System

LIST OF ACRONYMS

AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
BOP	balance of plant
CAP	corrective action program
CEDM	control element drive mechanism
CFR	*Code of Federal Regulations*
CRAI	condition report action item
CRDR	condition report disposition request
EDG	emergency diesel generator
ESFAS	engineered safety features actuation system
ET	eddy current test
FW	feedwater
HPSI	high pressure safety injection
ISI	inservice inspection
LER	licensee event report
LO	lube oil
LOP	loss of power
LOOP	loss of offsite power
MS	main steam
MSPI	mitigating systems performance index
NDE	non-destructive examination
NCV	noncited violation

NEI	Nuclear Energy Institute
NRC	*Nuclear Regulatory Commission*
OD	operability determination
PI	performance indicator
PLC	programmable logic controller
PREACS	pump room exhaust air cleanup system
PVAR	Palo Verde Action Request
PWSCC	primary water stress corrosion cracking
PT	dye penetrant test
QSPDS	quality safety parameter display system
RCS	reactor coolant system
RT	radiography test
SDC	shutdown cooling
SFP	spent fuel pool
SG	steam generator
SI	safety injection
SM	shift manager
SSC	structure, system, and component
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UT	ultrasonic test
VT	visual test
VUHP	vessel upper head penetration
WO	work order