ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-528; 50-529; 50-530
License Nos.:	NPF-41; NPF-51; NPF-74
Report No.:	50-528/97-25; 50-529/97-25; 50-530/97-25
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Road Tonopah, Arizona
Dates:	November 3-7, 1997
Inspectors:	D. Pereira, Reactor Inspector, Engineering Branch W. Wagner, Senior Reactor Inspector, Engineering Branch
Approved By:	Thomas F. Stetka, Acting Chief, Engineering Branch Division of Reactor Safety

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Palo Verde Nuclear Generating Station, Units 1, 2, and 3
NRC Inspection Report 50-528/97-25; 50-529/97-25; 50-530/97-25

During the period of November 3-7, 1997, two NRC inspectors conducted an inspection to followup issues previously identified in other NRC inspection reports.

Engineering

- One noncited violation was identified for the inability of the auxiliary feedwater system to automatically provide feedwater to the steam generators upon an auxiliary feedwater actuation signal under certain accident conditions (Section E8.2).

- One violation was identified for the failure to have adequate acceptance criteria for the inspection of the reactor coolant pump motor lubricating oil collection system flexible covers (Section E8.7).

Report Details

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Violation 50-528;-529;-530/9412-01 : This violation involved the failure of test procedures to incorporate appropriate acceptance criteria.

a. Background

This violation identified five examples of test procedures failing to incorporate acceptance criteria. In Response Letter 102-03091-WLS/AKK/DLK, dated August 23, 1994, the licensee disagreed with the second and third parts of the first example of the violation. The NRC reviewed additional information as documented in NRC Inspection Report 50-528;-529;-530/95-13, dated July 21, 1995, and concluded that the violations were valid as cited. In addition, the NRC concluded that corrective actions for the first part were sufficient to address the problems identified in the second and third parts. The NRC also concluded that no additional review of the first example of the violation was needed.

The licensee also disagreed with the second example of the violation. Specifically, the second example discussed the acceptance criteria in Surveillance Test 32ST-9ZZ03, "Surveillance Test Procedure for Class 4160 Bus Undervoltage Relays.". The licensee recognized that the setpoints specified in the technical specifications for the degraded voltage relays were not correct. As a result, the licensee had been administratively controlling the as-left degraded voltage relay setpoints at the high end of the acceptance criteria band. This issue, and the interim resolution, was the subject of a previous noncited violation (50-528/9135-01), in which the licensee was encouraged to continue the practice until the licensing basis for the setpoint was resolved. A technical specification amendment request had been approved, and Procedure 32ST-9ZZ03 was revised to incorporate the approved acceptance criteria. The licensee disagreed with the NRC's conclusion that administratively controlling the degraded voltage relay setpoints at the high end of the band, until the technical specification amendment request was approved, was an example of an additional violation of 10 CFR Part 50, Appendix B, Criterion XI. The NRC agreed with the licensee's conclusion. Additional information provided in the licensee response letter indicated that a historical review of the degraded voltage relay settings confirmed that the administrative controls were effective in ensuring that the relays were being properly set at the high end of the band.

b. Inspector Followup

The inspectors confirmed that the licensee's corrective actions were appropriate. Specifically, the licensee's corrective actions that were taken to address the specific violations were as follows:

- Procedure 39AC-9ZZ02, "Valve Services Maintenance," was revised to include the requirement that stroke-time acceptance testing must be considered following any motor pinion, worm shaft, or worm gear replacement (Example 1).

- Procedure 32MT-9ZZ56, "Motor Operator Valve Testing using MOVATS [Motor Operated Valve Actuation Testing System] Series 3000/3386 Systems," was revised. This revision required that as-found and as-left stroke times were compared and that valve services engineering review significant stroke-time differences and verify motor-operated valve stroke-time acceptability (Example 1).

- Procedure 32ST-9ZZ03, "Surveillance Test Procedures for the Class 4160 Bus Undervoltage Protective Relays," was revised with the approved technical specification amendment request acceptance criteria (Example 2).

- Procedures 43ST-3SP02, "Essential Spray Pond Pump Operability 4.0.5," and 43ST-3EW02, "Essential Cooling Water Pump Operability 4.0.5," were revised to include appropriate acceptance criteria and instrument uncertainty allowances. The revised procedures included clarification relative to the purpose of the tests (Example 3).

- Diesel Generator Loading Calculation 13-EC-MA-221 was reviewed to verify that the auto-connected loads were accounted for and did not exceed 5500 kW under worst-case accident conditions (Example 4).

- Procedure 73ST-3DG01, "Class 1E Diesel Generator and Integrated Safeguards Surveillance Test," was revised. The revised procedure contained appropriate acceptance criteria for verifying that the auto-connected diesel generator loads did not exceed 5500 kW (Example 4).

- Licensee Event Report 528/94-004-00, "Control Room Isolation Damper Not Reconnected Following Preventive Maintenance," was submitted to report a condition prohibited by the technical specifications.

The inspectors reviewed the revised procedures and the corrective actions taken to prevent recurrence. The inspectors concluded that the information regarding the reason for the violations, and the corrective actions taken to correct the violations were appropriate.

E8.2 (Closed) Licensee Event Report 50-528;-529;-530/96-19: Auxiliary Feedwater Beyond Component Level Design Basis.

a. Background

Licensee Event Report 95-013 documented an accident condition that exceeded the component level design basis for the auxiliary feedwater system. On December 1, 1995, the licensee determined that the auxiliary feedwater system was unable to perform the component-level design basis function of automatically providing water to the steam generators upon an auxiliary feedwater actuation signal (AFAS) under certain accident conditions. The licensee stated that these conditions existed for a limited range of main steam line break sizes coincident with a single failure of the motor-driven auxiliary feedwater (AFW) pump and below-normal steam generator levels. They considered this event to have a very low probability of occurrence (approximately 4E-12). Under these conditions, the initial AFAS would provide steam to the turbine-driven AFW pump from the faulted steam generator. When the faulted steam generator level and pressure decreased to zero, the turbine-driven pump steam admission valves would open fully and the turbine governor would increase its setpoint to approximately 3600 rpm while the turbine slowed to zero rpm due to a lack of steam. Under these conditions, the level in the intact steam generator would decrease due to a lack of auxiliary feedwater. The reduced steam generator level in the intact steam generator would cause a second AFAS to occur. This AFAS would allow steam from the unfaulted steam generator to be provided to the turbine-driven pump. However, due to the fact that the turbine-driven pump steam admission valves were full open and the governor set for maximum speed, there was high potential that the turbine-driven pump would trip on overspeed. The licensee stated that emergency operating procedures and operator action were capable of mitigating this event by resetting the turbine overspeed trip and/or starting the nonsafety motor-driven AFW pump from the control room.

The licensee performed an assessment to demonstrate that the existing condition did not pose additional safety concerns. In addition, the plant review board reviewed the event scenario and the assessment and determined that the postulated accident did not raise an unreviewed safety question. Based on recommendations from the plant review board, a justification for continued operation was prepared to support continued plant operation until permanent corrective action was implemented.

As permanent corrective action, a design change was installed in each unit. This design change modified the turbine-driven pump steam admission control by providing steam to the turbine from both steam generators simultaneously. The change precluded the turbine-driven AFW pump from tripping on overspeed because it eliminated the condition of suddenly suppling steam to the turbine with full open admission valves. The turbine now had a continuous steam supply under the postulated accident conditions.

In reviewing various licensing commitments, the licensee also determined that statements in the Updated Final Safety Analysis Report, regarding Combustion Engineering interface requirements did not conform to Updated Final Safety Analysis Report licensing commitments. Specifically, the Updated Final Safety Analysis Report Section 5.1.5.G.6, statement that, "The AFW system will deliver flow to the steam generators automatically upon receipt of an AFAS. . . ." was not correct for the postulated scenario.

b. Inspector Followup

The inspectors verified that the design changes were installed in all units via Design Modification DMWO 0074517. This was the design modification initiated to prevent the postulated AFW turbine-driven pump overspeed. The inspectors noted that the design change altered the auxiliary relay cabinet logic so that steam would be supplied to the turbine-driven AFW pump from both steam generators simultaneously. The inspectors verified that these logic changes would prevent the postulated overspeed trip and determined that the changes mitigated the postulated accident scenario.

The inspectors determined that a violation occurred when the licensee determined that the AFW system was not able to perform a component-level design basis function to automatically provide water to the steam generators upon an auxiliary feedwater actuation signal. The failure of the AFW system to perform its design function was identified as violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." However, the licensee identified this violation and took appropriate corrective action by performing appropriate safety evaluations and modifying the initiation logic on all three units. The violation was not a repeat of a previous violation and did not appear to be willful. This nonrepetitive, licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-528;-529;-530/9725-01).

E8.3 (Open) Inspection Followup Item 50-528;-529;-530/9719-03: Lack of safety evaluation for deletion of approximately 80 emergency commitments.

a. Background

This item was identified by the licensee and documented in Condition Report/ Disposition Request (CRDR) 9-7-Q257. The licensee found that approximately 80 commitments to the emergency plan had been deleted without a 10 CFR 50.59 safety evaluation being performed prior to revising the emergency plan procedures. The licensee planned to complete corrective actions for this finding by August 25, 1997.

b. Inspector Followup

The inspectors determined that the corrective actions of CRDR 9-7-Q257 had not yet been completed and that the approximately 80 commitments were actually 71 commitment items after eliminating the duplicate items. The inspectors found that the licensee was in process of verifying that reactivated commitments were correctly incorporated into the affected procedures and was in process of reverifying that all existing commitments were incorporated into procedures.

The licensee initiated a new CRDR, 9-7-O749 dated May 30, 1997. This CRDR described the potential that other commitments could have been inactivated during the procedure reduction process and that new procedures may not include all these commitments. The corrective actions for this CRDR were also still in progress during this inspection period.

During discussions with the licensee, the inspectors determined that additional corrective actions by other departments were still being implemented and decided. This followup item will remain open pending further NRC review of the corrective actions for CRDRs 9-7-Q257 and 9-7-0749.

E8.4 (Closed) Violation 50-528;-529;-530/9719-06: Failure to report six main steam safety valves out-of-technical specification tolerance.

a. Background

Prior to the Unit 3 sixth refueling outage, the main steam safety valves (MSSV) were tested using the Trevitest methodology to determine the as-found setpoints. As the result of this testing, the licensee documented that six of the MSSV as-found setpoints were outside the tolerance specified in Technical Specification 3.7.1, and that licensee personnel failed to report these out-of-tolerance conditions pursuant to 10 CFR 50.73. Subsequently, as a result of NRC questioning, licensee personnel performed a reportability evaluation and determined that the condition was reportable and discovered that there were seven MSSVs with as-found setpoints that exceeded the technical specification setpoint limit. The corrective actions included the issuance of Licensee Event Report 50-530/97-03 and a briefing of compliance personnel on this event.

b. Inspector Followup

The inspector verified that the licensee's corrective actions were completed and determined that those actions would prevent a recurrence. Specifically, the inspectors verified that the licensee issued Licensee Event Report 50-530/97-03 on July 7, 1997, and issued a letter to the CRDR review committee requesting that a reportability review be required for any instance where an MSSV or pressure relief valve fails to meet the technical specification tolerances.

E8.5 (Closed) Violation 50-528;-529;-530/9719-04: Inadequate corrective action that caused two containment spray system water hammer events.

(Closed) Inspection Followup Item 50-528;-529;-530/9719-05: Lack of surveillance requirements to ensure that the containment spray system was vented periodically.

a. Background

From July 21-26, 1995, the Unit 2 containment spray system experienced water hammer events during surveillance testing. As a result of the Unit 2 event, the licensee implemented changes in Procedure 40OP-9SI02, "Recovery from Shutdown Cooling to Normal Operating Lineup," to provide guidance regarding which containment spray valves were to be vented and when the venting was to occur. The licensee's investigation of the event indicated that air was entrapped on the discharge side of the pump.

On April 25, 1997, a water hammer event involving the Unit 3 containment spray system occurred. During the investigation of this event, the licensee determined that the procedural changes implemented as part of the corrective actions for the Unit 2 event, could be misinterpreted and result in the containment spray header not being vented. Specifically, the procedure changes required the venting of the containment spray system, if the safety injection system was being "restored from an outage/ maintenance condition." The licensee further determined that since containment spray surveillance testing could cause air entrapment, venting of the system after this testing was necessary. However, since Procedure 40OP-9S102 only required venting after an outage or maintenance condition, licensee personnel concluded that the containment spray piping venting was not required.

The licensee reviewed these two water hammer events, and concurred with the NRC finding that the corrective actions implemented as a result of the Unit 2 water hammer event were ineffective in preventing recurrence of a similar event. The licensee's investigation also determined that the use of conditional-type "if/then" statements was not the root cause of this event.

b. Inspector Followup

The inspectors reviewed the revised procedures for the containment spray system and the recovery from shutdown cooling to normal operating lineup procedure. The inspectors verified that the revised procedures would vent the containment spray system on a monthly basis. In addition, the inspectors verified that the following corrective actions were taken:

- Units 1, 2, and 3 procedures for containment spray valve verification were revised to require venting of the containment spray system on a monthly basis. This action was completed on June 27, 1997.

- Procedure 40OP-9SI02 was revised to require the venting of the containment spray system anytime there was a recovery from shutdown cooling. This action was completed on June 4, 1997.

- Additional procedures were reviewed to identify instances where similar conditional "if/then" statements were used. This review identified five procedures where such statements could cause confusion. These procedures were planned to be revised no later than April 18, 1998.

- Licensed and nonlicensed operations' staff were briefed on the importance of adequately venting systems following maintenance. This action was planned to be completed by November 30, 1997.

The inspectors concluded that the licensee's completed and planned actions would adequately resolve the violation.

E8.6 (Closed) Licensee Event Report 50-528/96-006: Cracked weld on high pressure safety injection minimum flow recirculation line forces Technical Specification Limiting Condition for Operation 3.0.3 entry.

a. Background

On October 29, 1996, Unit 1 was in Mode 3 at approximately 2250 psig and 565°F, when an operator identified a leak through a cracked weld in piping near Drain Valve 1PSIB024. This valve was on the minimum flow recirculation line for the Train B high pressure safety injection pump. Control room personnel entered Technical Specification Limiting Condition for Operation 3.0.3 following a determination that the statement for Technical Specification Limiting Condition for Operation 3.4.9 could not be met with reactor coolant system temperature greater than 210°F. At approximately 2:07 a.m., the Train B high pressure safety injection pump minimum recirculation line was isolated, Technical Specification Limiting Condition for Operation 3.4.9 was met and Technical Specification Limiting Condition for Operation 3.0.3 was exited. On October 30, 1996, the Unit 1 Train B minimum recirculation line was repaired and returned to service.

The apparent cause of the failure was incomplete fusion at the root of the socket weld that made it susceptible to crack initiation and propagation from vibration. Visual inspections of the remaining recirculation lines in Units 1, 2, and 3 were completed by the licensee and did not identify any leakage.

The licensee developed an action plan to perform vibration measurements for five different valve configurations on similar types of valves on each train in each of the three units, for a total of 30 valve locations. Vibration readings were taken during operation of

the high pressure safety injection pump to determine if the pump was causing the valve to vibrate at a frequency that might cause the weld to crack. However, of the 30 valve locations, a total of 6 of the locations, 2 in each unit, were inaccessible due to radiation levels, contamination levels, or because operations had to secure the pump prior to all readings being obtained.

b. Inspector Followup

The inspectors reviewed the weld repair documentation and determined that the repair and installation was conducted in accordance with the applicable requirements of the ASME Boiler and Pressure Vessel Code. The licensee performed sufficient nondestructive examinations before, during, and after the repair process to provide assurance that the pressure boundary was restored to within its original structural limits.

The inspectors also reviewed the vibration data results and the licensee's conclusions. The inspectors found that the licensee concluded that the vibration data indicated that vibration induced stress was not a cause of the cracked weld and confirmed that the cracked weld was due to incomplete fusion at the root weld. The inspectors agreed with the vibration data results and considered the corrective actions to be appropriate to prevent recurrence of this event.

E8.7 (Closed) Unresolved Item 50-528/9616-03: Degraded flexible covers for the reactor coolant pump motor lubrication oil collection system.

a. Background

The licensee performed inspections of the reactor coolant pump motor lubrication oil collection system in accordance with Procedure 31FT-9RC01, "RCP Lubrication Oil Collection System Inspection," Revision 2. On October 21, 1996, maintenance personnel performed Procedure 31FT-9RC01 as a prerequisite for a mode change on Unit 1. This procedure required an inspection of the flexible covers (silicon treated glass cloth shields) for the reactor coolant pump motor hydraulic power units. The personnel performing the inspection concluded that the covers were in good condition and properly installed. Four days after this inspection, the NRC found the flexible covers to be torn and improperly secured. The licensee initiated CRDR 9-6-1247 to evaluate the as-found condition of the flexible cover deficiencies.

The licensee initiated a work request to evaluate and repair these covers and subsequently replaced two of the covers, repaired the remaining two covers, and ensured that all the covers were properly fastened. Procedure 31FT-9RC01 was then reperformed to assure that the oil collection system was in accordance with design requirements prior to entering Mode 4.

b. Inspector Followup

The inspectors reviewed CRDR 9-6-1247. The CRDR included an evaluation to determine if the Unit 1 reactor coolant pump hydraulic power unit flexible covers would have met the design requirements in their as-found condition. The licensee concluded in the CRDR that the degraded condition of the covers was not serious and would not have compromised plant safety.

As the result of Unit 3 entering a refueling outage, the licensee initiated an additional CRDR, 3-7-0079, to perform an engineering evaluation to determine the status of these flexible covers and to determine if the four flexible covers on the Unit 3 reactor coolant pump hydraulic power units could perform their design basis function. On February 27, 1997, the licensee inspected the covers. While the licensee found that the condition of the covers was acceptable, they also found that the covers were secured to the hydraulic power units on only three sides with hold down clasps and that the remaining side of the covers was not secured due to the lack of hold down clasps. These covers were subsequently replaced under Work Order 777610. The licensee concluded that the as-found covers would perform their design basis function, which was to contain potentially leaking oil after a seismic event.

Since Unit 2 was in operation (which made the Unit 2 covers inaccessible for inspection) when the Unit 1 and Unit 3 findings were identified, the licensee evaluated whether continued operation of the unit was acceptable. Based on the evaluations performed for the Unit 1 (CRDR 9-6-1247) and Unit 3 (CRDR 3-7-0079), flexible covers, the licensee concluded that the Unit 2 flexible covers would perform their design basis function even with missing hold down clasps. Furthermore, the licensee concluded that the flexible cover as-found conditions for all the units was consistent with the UFSAR Appendix 9A description and, therefore, was not reportable.

An inspection of the Unit 2 flexible covers was performed by the licensee under Work Order 801238 on October 21, 1997, during the Unit 2 refueling outage. The covers were also found to be missing hold down clasps from one side. The flexible covers were replaced under Work Order 801238.

The inspectors reviewed Procedure 31FT-9RC01, Revision 2, and found that Section 8.3.10 of this procedure provided ambiguous instructions. These instructions did not provide sufficient inspection criteria such that personnel could determine what an acceptable or unacceptable flexible cover condition was because such items as material condition (e.g., holes or tears) and proper cover installation were not addressed. The licensee's investigation into this matter included interviews with personnel that inspected the Units 2 and 3 flexible covers. Interviews revealed that personnel had assumed that the unsecured side of the flexible covers were intentional because no clasps or fasteners were installed on the reactor coolant pump motor hydraulic power units and that, therefore, the installation was acceptable.

The licensee's corrective actions included repairing/replacing all of the flexible covers and revising Procedure 31FT-9RC01. The inspectors reviewed Revision 3 to this procedure and found that the procedure now included appropriate acceptance criteria to ensure that the material condition and installation of the flexible covers were defined. The inspectors concluded that the previous Revision 2 of Procedure 31FT-9RC01 did not provide the appropriate acceptance criteria required to perform a satisfactory inspection of the flexible covers.

10 CFR Part 50, Appendix B, Criterion V, states, in part, that instructions, procedures, or drawings shall include appropriate acceptance criteria for determining that important activities have been satisfactorily accomplished. Failure to include appropriate acceptance criteria in Procedure 31FT-9RC01 is a violation of 10 CFR Part 50, Appendix B, Criterion V (50-528/9725-02).

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on November 7, 1997. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Burns, Section Leader, Nuclear Regulatory Affairs Compliance
R. Fullner, Director, Nuclear Assurance Department
F. Gowers, Site Representative, EPE
R. Henry, Site Representative, SRP
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D. Marks, Section Leader, Nuclear Regulatory Affairs Compliance
M. Sontag, Section Leader, Nuclear Assurance Department
R. Younger, Operation Leader, Nuclear Assurance Department

NRC

J. Moorman, Senior Resident Inspector

LIST OF INSPECTION PROCEDURES USED

IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902:	Followup-Engineering

ITEMS OPENED AND CLOSED

Opened

50-528;-529;-530/9725-01	NCV	Auxiliary feedwater beyond design basis following steam line break
50-528/9725-02	VIO	Degraded reactor coolant pump flexible covers

Closed

50-528;-529;-530/9412-01	VIO	Inappropriate test acceptance criteria
50-528;-529;-530/9619-00	LER	AFW beyond component design basis following steam line break

50-528;-529;-530/9719-06	VIO	Failure to report six main steam safety valves out of technical specification tolerance
50-528;-529;-530/9719-04	VIO	Inadequate corrective action that caused two spray water hammer events
50-528;-529;-530/96-301	LER	Technical specification 3.0.3 entry due to cracked high pressure safety injection line weld
50-528/9616-03	URI	Degraded reactor coolant pump flexible covers
50-528/9725-02	VIO	Degraded reactor coolant pump flexible covers
50-528;-529;-530/9719-05	IFI	Lack of surveillance requirements to assure that containment spray was adequately vented
50-528;-529;-530/9725-01	NCV	Auxiliary feedwater system beyond component design basis following steam line break

<u>Discussed</u>

50-528;-529;-530/9719-03	IFI	Lack of safety evaluation for deletion of approximately 80 emergency plan procedure commitments