UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

August 6, 2001

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

SUBJECT: NRC INTEGRATED INSPECTION REPORT 50-528/01-03; 50-529/01-03; 50-530/01-03

Dear Mr. Overbeck:

On July 7, 2001, the NRC completed an inspection at your Palo Verde Nuclear Generating Station, Units 1, 2, and 3, facility. The enclosed report documents the inspection findings which were discussed on April 13, June 8, and July 3, 2001, with you and other members of your staff.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch D
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Dockets: 50-528
50-529
50-530
Licenses: NPF-41
NPF-51
NPF-74

Enclosure:
NRC Inspection Report
50-528/01-03; 50-529/01-03; 50-530/01-03

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ENCLOSURE

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:	50-528/01-03, 50-529/01-03, 50-530/01-03
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Road Tonopah, Arizona
Dates:	April 1 through July 7, 2001
Inspectors:	J. H. Moorman, III, Senior Resident Inspector N. L. Salgado, Resident Inspector G. G. Warnick. Resident Inspector M. P. Shannon, Senior Health Physicist B. D. Baca, Health Physicist W. C. Sifre, Senior Project Engineer J. E. Whittemore, Senior Reactor Inspector A. T. Gody, Senior Project Engineer
Accompanying Personnel:	G. F. Larkin, Reactor Engineer
Approved By:	Linda Joy Smith, Chief, Project Branch D Division of Reactor Projects
Attachment	Supplemental Information

SUMMARY OF FINDINGS

Palo Verde Nuclear Generating Station
NRC Inspection Report 50-528/01-03, 50-529/01-03, 50-530/01-03

IR 05000-528-01-03, IR 05000-529-01-03, IR 05000-530-01-03, on 04/01-07/07/01, Arizona Public Service Company, Palo Verde Nuclear Generating Station; Units 1, 2, and 3. Integrated resident and regional report. No findings identified.

The inspection was conducted by resident inspectors, regional senior project engineers, a regional health physicist, a regional senior health physicist, and a regional senior reactor inspector. No findings of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by No Color or by the severity level of the applicable violation.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Violations

Violations of very low significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Unit 1 was in Mode 3 for the start of the ninth refueling outage at the beginning of this period. The unit was restarted on May 12, 2001, and reached 100 percent power on May 18. The unit remained at 100 percent power for the remainder of this inspection period.

Unit 2 operated at 100 percent power for the duration of this inspection period.

Unit 3 operated at 99 percent power until May 19, 2001, when power was reduced to 20 percent for repairs to the main turbine and main condenser. While at 19 percent power, the unit tripped on a valid core protection calculator generated auxiliary trip signal of axial shape index less than (-)0.5. The unit was restarted on May 21 and reached 100 percent power on May 25. The unit remained at 100 percent power for the remainder of this inspection period.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed sections of the Updated Final Safety Analysis Report, the Design-Basis Manual, and other plant documents to determine if the condensate storage tank, refueling water tank, and essential spray pond systems were designed to remain functional during adverse weather related risks identified for the site. The documents reviewed are listed at the end of this report.

b. Issues and Findings

No findings of significance were identified.

1R04 Equipment Alignments - Routine Inspection (71111.04)

.1 Partial Walkdown Inspections

a. Inspection Scope

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

- Fuel pool cooling system (Unit 1)
- High pressure safety injection system Train A (Unit 2)
- Containment spray system Train A (Unit 2)

b. Findings

No findings of significance were identified.

.2 Complete Walkdown of the Emergency Diesel Generators and Diesel Fuel Oil Storage and Transfer Systems (Unit 3)

a. Inspection Scope

The inspectors completed a detailed alignment verification of the emergency diesel generators and fuel storage and transfer systems. This verification included a review of Design-Basis Manual, Updated Final Safety Analysis Report, Procedure 40OP-9DF01, "Diesel Fuel Oil Storage and Transfer," Revision 25, Procedure 40OP-9DG01, "Emergency Diesel Generators," Revision 16, applicable plant drawings, outstanding modifications, work orders, operator workarounds, and CRDRs. The inspectors verified the following:

- All valves were properly aligned
- There was no leakage that could affect operability
- Electrical power was available as required
- Control room and local instrumentation and controls were properly aligned
- Major system components were properly labeled and lubricated
- Jacket waters were properly aligned and functioning
- Support systems were properly aligned
- Hangers and supports were correctly installed and functional

The inspectors also verified that the licensee was identifying and documenting equipment alignment problems at an appropriate threshold in the corrective action program.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Monthly Routine Inspection

a. Inspection Scope

The inspectors conducted tours of the areas listed below that are important to reactor safety and referenced in the Prefire Strategies Manual to evaluate conditions related to licensee control of transient combustibles and ignition sources; the material condition, operational status, and operational lineup of fire protection systems, equipment, and features; and the fire barriers used to prevent fire damage from propagation of potential fires.

- Reactor Building - all accessible elevations (Unit 1)
- Reactor Building 100-foot elevation - numerous grinding and welding jobs (Unit 1)
- Auxiliary Building 100-foot elevation (Unit 2)
- Control Building 74-foot and 100-foot elevations (Unit 3)
- Gas turbine generator facility
- Auxiliary Building 120-foot and 140-foot elevations (Unit 1)

b. Findings

No findings of significance were identified.

.2 Fire Drill - Emergency Diesel Generator Building A (Unit 2)

a. Inspection Scope

On April 9, 2001, the inspectors observed a fire brigade drill from the control room and locally, to evaluate the readiness of the licensee's personnel to prevent and fight fires. The inspectors reviewed the strategies and information in the Prefire Strategies Manual, Revision 13, to verify that it accurately described the fire protection design features, fire area boundaries, and combustible loading for emergency diesel generator Building A. The inspectors observed fire brigade personnel enter the fire area and utilize the prefire plan strategies. The inspectors observed the equipment brought to the scene to evaluate whether sufficient equipment was available for the simulated fire. The inspectors observed fire fighting directions and radio communications between the fireground commander, fire department personnel, and the control room. Finally, the inspectors observed the post drill critique to evaluate if the drill acceptance criteria was satisfied.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, the Design-Basis Manual and other licensee documents to verify that the internal flood mitigation plans and equipment were consistent with the plants' design requirements and risk analysis assumptions. The inspectors also conducted walkdowns in Units 1, 2, and 3 auxiliary feed pump rooms and lower elevations of the control buildings, auxiliary buildings, and the main steam support structures for susceptibility to internal flooding. The inspectors verified the positions of valves as required to prevent internal flooding caused by system draindowns. The inspectors also verified the status of the floor drain check valves, sump room level detection equipment and associated alarm circuitry, and the integrity of walls, ceilings, and piping penetration seals.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On May 22, 2000, the inspectors observed licensed operator simulator training. The inspectors evaluated licensed operator performance for adherence to principles of sound reactor plant operation. The inspectors evaluated the evaluators critique to determine the depth of the evaluation and extent of feedback given to the operators. The inspectors evaluated the training scenario to determine if it was challenging and relevant to the operators' training needs.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

.1 Effectiveness of Current Maintenance Rule Program Activities

a. Inspection Scope

The inspectors reviewed the licensee's maintenance rule program documented in Procedures 70DP-OMR01, "Maintenance Rule," Revision 5; and 90DP-OIP10, "Condition Reporting," Revision 11, to confirm that the integration of these two procedures implemented the requirements of 10 CFR 50.65 at the Palo Verde Nuclear Generating Station

The inspectors reviewed CRDRs, maintenance rule functional failure determinations, and Category (a)(1) recovery schemes. The bulk of the items sampled were for issues that affected the maintenance rule program functions of the structures, systems, and components (SSCs) related to the emergency diesel generators, reactor coolant system, and the high pressure safety injection system in the three units. For the identified systems and related functions, the inspectors assessed the maintenance rule program performance related to:

- Adequacy of cause determination for degraded performance or failure to meet performance criteria

- Adequacy of corrective actions

- Adequacy of established goals and subsequent monitoring of functions placed in Category (a)(1)

- Adequacy of program revisions to scoping, risk significance, performance criteria, and monitoring of the subject SSCs.

- The creation of new risk-significant functions to improve performance monitoring

The inspectors conducted interviews with the licensee's engineering and maintenance personnel to gain an understanding of actions taken with regard to specific issues. The inspectors also evaluated system engineers knowledge of the maintenance rule program electronic data base and programmatic knowledge related to performance criteria and monitoring periods.

b. Findings

No findings of significance were identified.

.2 Periodic Assessment Reviews

a. Inspection Scope

The inspectors reviewed licensee reports documenting the performance of the two most recent maintenance rule program periodic assessments for each unit. These periodic assessments are conducted to meet the requirements of 10 CFR 50.65(a)(3). The assessments covered the periods and were issued on the dates indicated in the table below:

ASSESSMENT PERIOD	ISSUE DATE
MARCH 1998 - AUGUST 1999	JANUARY 14 , 2000
SEPTEMBER 1999 - DECEMBER 2000	APRIL 17, 2001

The inspector's review was conducted to determine if the reports contained adequate assessment of the performance of the maintenance rule program, as well as conformance with applicable programmatic and regulatory requirements. To accomplish this, the inspectors examined the licensee's staff evaluative efforts for the following elements in the reports:

- The program treatment of nonrisk-significant SSC functions monitored against plant level performance criteria

- Program adjustments made in response to unbalanced reliability and availability of risk-significant SSCs
- The application of industry operating experience
- Performance review of Category (a)(1) systems
- Evaluation of the bases for system category status change, e.g., Category (a)(1) to (a)(2) or Category (a)(2) to (a)(1)
- Effectiveness of performance and condition monitoring at component, train, system, and plant levels

The inspectors also verified that the issuance of the two most recent assessments met the regulatory timeliness requirements.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

Through the review and examination of CRDRs, root cause analyses, goal setting, and Category (a)(1) recovery schemes, the inspectors evaluated the use of the licensee's corrective action program as it pertained to the maintenance rule program. In addition, system engineers were interviewed to gain understanding of the engineers' performance in relation to specific SSC issues or problems. The purpose of this review was to establish that the corrective action program was entered at the appropriate threshold and effectively utilized for the purposes of:

- The initiation of appropriate action upon exceeding SSC performance trigger values that were set prior to performance criteria
- Starting the cause evaluation and determination of appropriate corrective action when performance criteria or condition limits were exceeded
- Identifying and correcting performance-related issues or conditions identified during the periodic assessment
- Identifying and correcting specific and common issues or conditions brought to light through activities such as performance trending or data analysis

Through interviews, the inspectors also examined the licensee staff's identification of issues and implementation of corrective action in support of the licensee's maintenance rule program. The inspectors further examined the corrective action program to determine if it was sufficiently integrated with the maintenance rule program to identify programmatic issues or weaknesses.

b. Findings

No findings of significance were identified.

.4 Routine review of Maintenance Rule Implementation

a. Inspection Scope

The inspectors evaluated the following equipment failures to verify that licensee personnel properly implemented the requirements of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”:

- Power supply failure caused spurious trip of several plant protection system Channel B parameters (Unit 2)
- Essential chiller Train B tripped on low refrigerant temperature (Unit 1)
- Emergency diesel generator Train B scored cylinder liner (Unit 3)
- Essential chilled water Train B pressure relief valve lifted lower than the allowed pressure during surveillance testing (Unit 1)
- Excess back-leakage through Train B, high pressure safety injection pump discharge Check Valve 2PSIBV405. One licensee identified violation related to this failure is documented in Section 4OA7 (Unit 2)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Throughout the inspection period, the inspectors reviewed daily and weekly work schedules to determine when risk-significant activities were scheduled. The inspectors reviewed selected activities regarding risk evaluations and overall plant configuration control to verify compliance with 30DP-9MT03, “Assessment and Management of Risk When Performing Maintenance in Modes 1 - 4,” Revision 1. The inspectors discussed emergent work issues with work control personnel and reviewed the potential risk impact of these activities to verify that the work was adequately planned, controlled, and executed. The specific activities reviewed were associated with planned and emergent maintenance on:

- Replacement of control element assemblies (Unit 1)

- Replacement of steam generator feed water piping (Unit 1)

- Repair of heated-junction-thermocouple guide tubes (Unit 1)

- Repair of instrument air Compressor A (Unit 3)

- Procedure 36ST-9SA01, “ESFAS Train A Subgroup Relay Functional Test,” Revision 22 (Unit 1)

- Troubleshoot unexpected transfer of Inverter PNB-N12 to alternate power supply (Unit 3)

- Emergency diesel generator, essential spray pond, essential chilled water, essential cooling water, and containment spray Train A online outage (Unit 2)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

.1 Reactor Trip From 20 Percent Power Due to Axial Shape Index Out of Band (Unit 3)

a. Inspection Scope

On May 19, 2000, Unit 3 reactor power was reduced to 19 percent to allow corrective maintenance on the main condenser and preventative maintenance on the main turbine. At 3:06 a.m., the reactor tripped from 19 percent power when axial shape index (ASI) values exceeded the trip setpoint in the core protection calculators. The excessive ASI values were caused by mild axial offset anomaly in the reactor core. The inspectors reviewed the unit logs and data taken from the trip to determine if operator response was in accordance with applicable procedures.

Unit 3 was restarted on May 20. The inspectors observed portions of the startup to determine if plant operations were conducted in accordance with approved procedures and within Technical Specification limits. One licensee identified violation that occurred during the startup is documented in Section 4OA7.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- CRDR 2375404, evaluation of control element assembly anomalies, including assembly 22 failing to fully insert into the core, observed during the planned reactor trip for Refueling Outage U1R9 (Unit 1)

- Operability Evaluation for aged control element assemblies in Unit 2 and Unit 3. This operability evaluation assessed the remaining lifetime of control element assemblies in Unit 2 and Unit 3 due to cracking problems experienced in Unit 1. The inspectors also reviewed the licensee's evaluation of updated results from the control element assembly lifetime limit computer model, empirical data from plant operations and testing, and design differences between Unit 1 and the other two units as documented in the Monday, April 9, 2001, 7:44 pm log entry into the Unit 2 and Unit 3 Unit Logs (Units 2 and 3)

- DFWO 2383464, evaluation to continue with U1R9 core reload with leaking reactor cavity refueling pool seal (Unit 1)

- Entry into Technical Specification 3.3.1, Condition A, upon mode 2 entry due to failed Temperature Element RCCTE112HC (Unit 1)

- Night order dated June 7, 2001. This night order discussed the application of Technical Specifications to the core protection calculator operating bypass when a log power channel has failed low (Units 1, 2, and 3)

- CRDR 2403932, evaluation to justijy compliance with Technical Specification 3.6.1 during containment pressure transmitter replacement (Unit 2)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Review of the Cumulative Effects of Operator Workarounds

On May 31 and June 1, 2001, the inspectors reviewed the control room deficiency log, night order book, temporary modification log, and procedure variance log in Units 1, 2, and 3, to determine the number of operator workarounds that existed. The inspectors also interviewed plant operators to assess the completeness of the logs reviewed.

Existing workarounds were assessed to determine if the cumulative effect could increase initiating event frequency or adversely affect the plant operators' ability to mitigate an accident.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and/or evaluated the results from the following postmaintenance tests to determine whether the test adequately confirmed equipment operability. The inspectors also verified that postmaintenance tests satisfied the requirements of 30DP-9WP04, “Postmaintenance Retest Development,” Revision 12.

- Work Order 2321785 — Retest performed following EDG Train B outage work (Unit 1)
- Work Order 2377088 — Replacement of Unit 1 full length control element assemblies (Unit 1)
- Work Order 2381739 — Replace 16-inch to 14-inch reducer on Steam Generator 2 economizer feed line (Unit 1)
- Work Order 2387797 — Auxiliary feedwater flow transmitter to steam generator 1 (Unit 1)
- Work Order 2396425 — Repair of oil leak on auxiliary feedwater Pump A governor (Unit 1)
- Work Order 2399041 — Troubleshoot unexpected transfer of Inverter PNB-N12 to alternate power supply (Unit 3)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Review of the Unit 1 Outage Plan

a. Inspection Scope

The inspectors reviewed the licensee's outage risk assessment, Palo Verde Unit 1 ninth refueling shutdown risk assessment, to verify that the licensee appropriately considered risk in planning and scheduling the outage activities.

The inspectors primarily focused on the following activities:

- Transition and midloop operation
- Fuel offload and reload

b. Findings

No findings of significance were identified.

.2 Monitoring of Shutdown Activities

a. Inspection Scope

The inspectors reviewed plant data records, control room and unit logs, and conducted interviews with licensed operators to assess the licensee's compliance with Technical Specification plant cooldown limits during the Unit 1 plant cooldown.

b. Findings

No findings of significance were identified.

.3 Control of Outage Activities

a. Inspection Scope

The inspectors reviewed plant conditions and observed selected refueling outage activities throughout the outage to verify that the licensee maintained the plant in a configuration consistent with the requirements of Technical Specifications and with the assumptions of the outage risk assessment. The inspectors verified that emergent issues were properly assessed for their impact on plant risk.

Electrical power availability was periodically verified to meet Technical Specification requirements and outage risk assessment recommendations. Control room operators were interviewed to determine if they were cognizant of plant conditions. The inspectors reviewed equipment clearance activities, controls for reactivity management, and reactor coolant system inventory.

b. Findings

No findings of significance were identified.

.4 Clearance Activities

a. Inspection Scope

The inspectors reviewed the following equipment clearances which established boundaries associated with risk-significant maintenance or modifications. The inspectors verified that clearance tags were hung on the proper equipment, that the equipment was properly configured to support the work performed, and that the requirements of 40DP-9OP29, “Permit and Tagging Process,” Revison 16, were satisfied.

- ID# 26346, ensure proper control of valves that could effect SFP inventory
- ID# 35679, PCNV-118 status control permit
- ID# 40727, installation of temporary plant cooling water to nuclear cooling water
- ID# 34770, to examine diesel engine general tear down and inspections
- ID# 40790, perform emergency core cooling water Train B sump inspection
- ID# 40805, safety injection Tank 2A outage permit
- ID# 40806, safety injection Tank 2B outage permit
- ID# 46066, safety injection Tank 1A outage permit

b. Findings

No findings of significance were identified.

.5 Reduced Inventory and Midloop

a. Inspection Scope

The inspectors observed, in part, Unit 1 midloop activities to verify that the licensee had appropriately considered the risk associated with this activity. The inspectors reviewed the licensee’s response to Generic Letter 88-17 and verified that licensee commitments had been properly translated into procedures. The inspectors also verified that multiple sources of electrical power, multiple reactor vessel level indications, and multiple reactor coolant system temperature indications were available. The inspectors observed licensee compliance with the following procedures:

- 40OP-9ZZ16 “RCS Drain Operations,” Revision 23
- 40OP-9ZZ20 “Reduced Inventory Operations,” Revision 3

b. Findings

No findings of significance were identified.

.6 Refueling Activities

a. Inspection Scope

The inspectors observed portions of core offload and core reload activities to determine if these activities were conducted in accordance with the Technical Specifications and administrative procedures. The inspectors observed licensee compliance with Procedure 72IC-9RX03, “Core Reloading,” Revision 15.

b. Findings

No findings of significance were identified.

.7 Monitoring of Heatup and Startup Activities

a. Inspection Scope

The inspectors performed control room observations and reviewed unit logs to verify that the Unit 1 startup was conducted in compliance with Technical Specifications, license conditions, and administrative requirements. The inspectors conducted a walkdown of all accessible areas of the Unit 1 containment building to assess containment cleanliness and material condition of components. Portions of reactor physics testing were observed and a review of Procedure 72PY-9RX04, “Low Power Physics Tests using RMAS,” Revision 0, was completed to verify that core operating limit parameters were consistent with the design.

b. Findings

No findings of significance were identified.

.8 Identification and Resolution of Problems

a. Inspection Scope

The inspectors screened CRDRs that documented problems identified during the Unit 1 outage to verify that problems were identified at an appropriate threshold and in accordance with the guidance in Procedure 90DP-0IP10, “Condition Reporting,” Revision 11.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed the performance of and/or reviewed documentation for the following surveillance tests. Applicable test data was reviewed to verify whether they met Technical Specification, Updated Final Safety Analysis Report, and licensee procedure requirements. Also, the inspectors verified that the testing effectively demonstrated that the systems were operationally ready, capable of performing their intended safety functions, and that identified problems were entered into the corrective action program for resolution.

- 73ST-9DG02, "Class 1E Diesel Generator and Integrated Safeguards Test Train B," Revision 2 (Unit 1)

- 73ST-9CL01, “Containment Leakage Type “B” and “C” Testing,” Revision 13 (Unit 1)

- 73ST-9DG01, “Class 1E Diesel Generator and Integrated Safeguards Test Train A," Revision 1, Section 8.6 (Unit 1)
- 72ST-9RX02, “Moderator Temperature Coefficient At Power,” Revision 13 (Unit 3)
- 73ST-9SI10, “HPSI Pumps Miniflow - Inservice Test,” Revision 19 (Unit 3)
- 72ST-9RX02, “Moderator Temperature Coefficient At Power,” Revision 14 (Unit 1)
- 40ST-9HF02, “Fuel Building Essential Ventilation System Operability Test,” Revision 11 (Unit 2)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors evaluated the following temporary modifications and associated 10 CFR 50.59 screenings against the system design-basis documentation, and verified that the modifications did not adversely affect system operability or availability. Additionally, the inspectors verified that the installations were consistent with applicable modification documents and conducted with adequate configuration control.

- T-Mod 2-99-PW-001 Installation of portable cooling towers to provide cooling to one of the nuclear cooling water heat exchangers during plant cooling water system outage (Unit 1)
- DFWO 2388545 Substitute the output of resistance temperature detector (RTD) 111X for 112HC (Unit 1)

b. Findings

No findings of significance were identified.

2. Radiation Safety
Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Controls to Radiologically Significant Areas (7112101)

a. Inspection Scope

Radiation workers and radiation protection personnel were interviewed concerning their radiation protection work requirements. A number of tours of the radiologically controlled area were conducted. The following items were reviewed and compared with regulatory requirements:

- Quality Assurance Evaluation Reports: ER 01-0039, ER 01-0076, ER 01-0099
- Access controls, surveys and radiation exposure permits for the following three significant high dose work areas in the radiologically controlled area: Steam Generator Maintenance (Radiation Exposure Permit 1-3306C), Reactor De-Stack and Re-Stack (Radiation Exposure Permit 1-3002C), and Control Element Assembly Replacement (Radiation Exposure Permit 1-3000A)
- Specified electronic dosimeter setpoints
- High radiation key controls
- Placement of personnel dosimetry
- Job coverage by radiation protection personnel
- Controls involved with the storage of highly radioactive items in the spent fuel pool
- Associated program procedures
- A summary of operational related radiation protection corrective action reports written since January 1, 2001, (twelve of these reports were reviewed in detail: CRDR 2351080, CRDR 2351173, CRDR 2352218, CRDR 2353777, CRDR 2356818, CRDR 2357496, CRDR 2358013, CRDR 2359534, CRDR 2360600, CRDR 2364818, CRDR 2367797, and CRDR 2371546

No job-in-progress reviews were required to be performed during the inspection week; therefore, this aspect of the procedure could not be verified.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events Cornerstone

a. Inspection Scope

The inspectors reviewed unit logs, plant thermal performance record, monthly operating reports, and licensee event reports from July 2000 to March 2001 to verify the accuracy and completeness of data used to calculate and report the following performance indicators:

- Unplanned scrams per 7000 critical hours
- Scrams with loss of normal heat removal
- Unplanned transients per 7000 critical hours

b. Findings

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

.2 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors reviewed a sample of unit logs, maintenance rule unavailability tracking data and technical specification component condition records from June 2000 to June 2001 to verify the accuracy and completeness of data used to calculate and report high pressure safety injection system unavailablity.

b. Findings

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

.3 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed corrective action program records for high radiation areas, locked high radiation areas, and unplanned exposure occurrences for the past four months to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area exit transactions with exposures greater than 100 millirem for the past quarter were reviewed. Selected examples were investigated to determine whether they were within the dose projections of the governing radiation exposure permits.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented during the past quarter to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Findings related to identification and resolution of problems are discussed in Sectiona 4OA3.3, 4OA7.4, and 4OA7.5.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 50-528;529;530/2000-004-00: Technical Specification Violation Due to Deficient Test Procedure for Refueling Purge Valves. The inspectors reviewed the LER and no findings of significance were identified. This Technical Specification violation was placed in the licensee's corrective action program and documented on Condition Report/Disposition Request 2342097. This event constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the Enforcement Policy.

.2 (Closed) Licensee Event Report (LER) 50-528/2001-001-00: Reactor Coolant System (RCS) Pressure Boundary Leakage Caused by Degraded Alloy 600 Instrument Nozzle. The inspectors reviewed the LER and determined the degradation had very low safety significance because the crack was small and axial. Axial cracks typically result in only small leaks. This event has been addressed and corrected through the licensee's corrective action program and documented on Condition Report/Disposition Request 2375490.

.3 (Closed) Licensee Event Report (LER) 50-529/2000-005-00;-005-01: Safety Injection Discharge Check Valve Back-Leakage Causes Degraded Safety Injection Flow.

a. Inspection Scope

On October 23, 2000, while PVNGS was in Mode 6, the licensee identified that the Unit 2, Train B, high pressure safety injection (HPSI) pump discharge check valve

(2PSIBV405) had excessive back-leakage beyond the acceptance criteria of 10 gpm at 50 psid contained in Licensee Procedure 73ST-9XI33, “HPSI Pump and Check Valve Full Flow Test.” Actual test results indicated that Valve 2PSIBV405 had a back-leakage flow rate of 30 gpm at a D/P of 12 psid. The Train B check valve was declared inoperable, repaired, and returned to service by November 3, 2000.

The inspectors:

- reviewed this LER
- reviewed associated CRDR 2332280
- corrective actions completed for similar 1997 and 1998 failures
- Borg-Warner check valve operating history at PVNGS
- interviewed plant engineering and maintenance personnel
- observed a demonstration of revised maintenance practices
- reviewed technical adequacy and timeliness of changes to Technical Manual Procedure 31MT-9ZZ17, “Disassembly and Assembly of Borg-Warner Check Valves, “ Revisions 4 through 11
- discussed check valve performance issues with the vendor

to confirm:

- the root cause was correctly identified
- the extent of the condition was correctly bounded
- planned and completed corrective actions had been prioritized appropriately
- system monitoring was being implemented as required by 10 CFR 50.65 a(1)
- 10 CFR Part 21 reportability requirements were met

b. Findings

Root Cause

The licensee’s root cause determination revealed that the check valve disk had become lodged in the seat because of excess articulation in one orientation (about 20 degrees of rotation). The excess articulation was caused by a manufacturing defect (uneven weld buildup on disk to stud attachment point) which resulted in variable articulation dependant upon the rotational position of the disk in relation to the swing arm. Check Valve 2PSIBV405 is a safety-related American Society of Mechanical Engineer (ASME) Class 2, 4-inch, 1500-pound, bonnet pressure seal, two piece welded body swing check valve manufactured by Borg-Warner (BW) (Drawing Number 79120).

The licensee sent a letter to Flowserve (Current owner of Borg-Warner) on May 31, 2001, discussing the apparent manufacturing defect. The inspector also contacted the vendor and discussed their inspection, which was ongoing at the time. The vendor noted that a similar manufacturing defect was the subject of a February 12, 1993, 10 CFR Part 21 report. However, the scope of the 1993 report was restricted to Borg-Warner 4-inch, 150-pound, bolted-bonnet, swing check valves and would not have been applicable to the 4-inch, 1500-pound valve discussed above. Flowserve concluded the potential scope of the problem should be expanded to include all Borg-Warner 3-inch and 4-inch swing check valves and replacement disk assemblies that were furnished in 1977 or earlier. Flowserve planned to notify their customers of the change in scope.

Prior Corrective Actions Ineffective

These check valves had a history of excess back-leakage and were the subject of an NRC Special Inspection completed on July 28, 1998 (NRC Inspection Report 50-528/529/530;98-14). Following the issues identified prior to mid-1998, the licensee implemented a number of corrective actions involving correcting maintenance and surveillance procedures. The errors identified in these procedures involved inadequate acceptance criteria for emergency core cooling system forward flow testing, check valve leakage acceptance criteria, and maintenance practices which failed to ensure proper vertical alignment and articulation of the disk in the subject BW check valves. Maintenance procedure changes in 1998 added a requirement to measure disk articulation but did not require the mechanic to rotate the disk and record the maximum and minimum articulation values. The licensee identified that their prior corrective actions were not effective at precluding recurrence of the problem. See Section 4OA7.5 for enforcement related to this issue.

Significance Determination

The inspectors used the significance determination process to evaluate the significance of the excessive back-leakage. To support this determination the inspectors and the senior reactor analyst considered the duration of the condition, the extent of the condition, and the likelihood of successful operator recovery. The inspectors also reviewed hydraulic and emergency core cooling analyses that were used by the licensee to assess the realistic core damage potential and conferred with the Office of Nuclear Reactor Regulation to determine whether these realistic model assumptions were reasonable. A summary of the licensee's realistic assumptions is listed below.

1. **Emergency Core Cooling Analysis of Posttrip Main Steam Line Break: Two initial conditions were considered to determine if a return to power would occur: (1) Hot Zero Power [HZP] and (2) Hot Full Power [HFP].**

   CESEC-III Code, Revision C89300Mod4, 10/21/97, was used. The calculation concluded no return to power for both the HZP and HFP cases. The inspector found that the assumptions were similar to the licensee’s analysis of record and were conservative.

In the HFP case, HPSI injection never occurred and reactivity was turned by moderator heating when the steam generators went dry. In the HZP case, HPSI injection occurred (between 200-300 gpm) as reactor coolant system pressure dropped, and the reactivity was turned due to the combination of moderator heating and the boration from HPSI injection.

2. Emergency Core Cooling Analysis of Small Break Loss of Coolant Accident: To determine if 10 CFR 50.46 (Appendix K) safety criteria would be met.

The licensee used parts of the ABB SBLOCA Realistic Evaluation Model, which has not been approved by the NRC as a design-basis model. The NRC's review was stopped after the vendor withdrew the report after a number of questions by the staff. The staff's concerns were primarily associated with the way the vendor dealt with uncertainty in the calculation. The following summarizes what the licensee modified from their Appendix K analyses of record:

(a) Initial power level was reduced from 105 percent to 102 percent. 10 CFR Part 50, Appendix K, requires a margin of 2 percent. The additional 3 percent margin added by the licensee in the original analyses was added for future contingencies. Although the 3 percent additional margin was included in their analysis of record, it was not needed to meet the Appendix K requirements and therefore was additional margin that could be removed.

(b) Rather than use the 1971 American Nuclear Society (ANS) decay heat model in the blowdown portions of the analyses in their analyses of record, the licensee used the 1979 ANS decay heat model. The primary difference in these models is that the 1971 model is based on data with a 2σ scatter of 20 percent and the 1979 model is based on a 2σ scatter of 6.3 percent. This decay heat model was used in the realistic evaluation models for another vendor. The staff found that the 1979 ANS Decay Heat Model conservatively modeled core decay heat and was therefore acceptable.

(c) The hot rod heatup portion of the analyses used in the ABB SBLOCA Realistic Evaluation Model has not been reviewed and approved by the NRC. Some of the attributes of this model used by the licensee in their realistic evaluation include:

(1) the 1979 ANS decay heat model described above,

(2) the Cathcart-Pawel high temperature and Biederman low temperature cladding oxidation models (Appendix K specifies the Baker-Just model), and

(3) the strain-to-failure model for determining cladding rupture.

The staff did not identify any concerns with the models used by the licensee as a better estimate of successful core cooling for their risk assessment. 10 CFR 50.46 allows a licensee to use alternate means to assess the potential for core damage as long as they meet the reporting criteria and develop adequate corrective actions. In this case, the licensee's alternate assessment concluded that the existing condition would not have resulted in core damage.

3. Hydraulic Model of As-found Conditions:

The licensee developed a model to predict the altered pump flow curves assuming the stuck open HPSI check valve acted as an orifice. Given the orientation of the check valve and the direction of reverse flow, the assumption that the check valves would act as if they were an orifice was conservative. The licensee's hydraulic analysis which was documented in Calculation 13-MA-SI-982, Revision 0, ". . . Assessment of as-found Leakage for 2PSIB-405 / 1PSIA-V-404 . . ." was found to be straight forward, clear, and conservative.

The issue was determined to be of very low safety significance because: (1) the more realistic assumptions were determined to be acceptable for this purpose and (2) when the potential for core damage was assessed using realistic modeling assumptions, the licensee concluded that the core would have been adequately cooled, during the most severe accidents.

4OA5 Exit Meeting Summary

The regional senior health physicist presented inspection results to Mr. G. Overbeck, Senior Vice President - Nuclear, and other members of licensee management at the conclusion of the inspection on April 13, 2001.

The regional engineering inspectors presented results of the maintenance rule biennial inspection activities to Mr. G. Overbeck, Senior Vice President - Nuclear, and other members of licensee management at the conclusion of the inspection on June 8, 2001.

The resident inspectors presented the inspection results to Mr. G. Overbeck, Senior Vice President - Nuclear, and other members of licensee management on July 3, 2001.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was provided to and reviewed by the inspectors. Arrangements were made at the resident inspector exit for final disposition of the proprietary material. No proprietary information is contained in this report.

4OA7 Licensee-Identified Violations - The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as noncited violations (NCV).

If the Palo Verde Nuclear Generating Station contests 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 Region IV; 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.

	NCV Tracking Number	Requirement Licensee Failed to Meet
.1	NCV 50-528/01-03-01	Technical Specification 5.4, "Procedures," requires that written procedures be implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 3.a of this Regulatory Guide includes instructions for draining the RCS. Procedure 40OP-9ZZ16, "RCS Drain Operations," Revision 24, provides instructions for draining the RCS and includes instrumentation requirements for reactor water level monitoring. On May 7, 2001, this procedure was being implemented to maintain Unit 1 in a partially drained condition. Due to personnel error involving plant configuration control, required reactor level instrumentation was isolated for a period of time as described in CRDR 2385849 and is being treated as a NCV. This finding is of very low safety significance (Green) because reactor water level was maintained within the operating band at all times.
2.	NCV 50-530/01-03-02	Technical Specification 5.4, "Procedures," requires that written procedures be implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 2.b of this Regulatory Guide includes instructions for reactor plant startup. Procedure 40OP-9ZZ03, "Reactor Startup," Revision 2, provides instructions for monitoring anticipated critical position during a reactor startup. On May 19, 2001, this procedure was implemented to conduct a reactor startup of Unit 3. Due to personnel error, the reactor startup was allowed to continue after two consecutive anticipated critical positions indicated that the Unit 3 reactor would go critical below the (-)500 pcm position as described in CRDR 2391526. This finding is of very low safety significance (Green) because criticality did not occur below Technical Specification limits.
3.	NCV 50-529/01-03-03	Technical Specification 5.4, "Procedures," requires that written procedures be implemented and maintained

covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 3.i of this Regulatory Guide includes instructions for operating the main steam system. Procedure 40OP-9SG01, “Main Steam,” Revision 23, provides instructions for operation of steam traps when they are removed from service. On March 7, 2001, Unit 2 Auxiliary Feedwater Pump AFA-P01 was made inoperable for approximately 3.5 hours after Steam Trap SGN-M23 was removed from service and not operated as described in 40OP-9SG01. This finding is described in CRDR 2369601. This finding is of very low safety significance (Green) because it did not cause a loss of the auxiliary feedwater safety function.

4. NCV 50-529/01-03-04

10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action requires that conditions adverse to quality be promptly identified and corrected. Procedure 90DP-0IP10, “Condition Reporting,” Revision 11, requires that a condition report be written for conditions such as a valve or component misalignment. On October 31, 2000, the operations crew that performed Procedure 73ST-9XI 33, “HPSI Pump and Check Valve Full Flow Test,” Revision 19, discovered a valve misalignment and did not document the misalignment in a condition report. This event was discovered during the investigation for CRDR 2332280, “Failure of HPSI Pump Discharge Check Valve 2PSIBV405,” and was determined to have caused a hydraulic transient in the high pressure safety injection system. This finding is described in CRDR 2352119. This finding is of very low safety significance (Green) because the hydraulic transient in the high pressure safety injection system did not damage the system.

5. NCV 50-529/01-03-05

10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires that the cause of significant conditions adverse to quality be determined and corrective action be taken to preclude repetition. Contrary to this, the licensee determined that they had not adequately addressed prior Borg-Warner check valve failures that occurred in 1997 and 1998, which were identified as significant conditions adverse to quality. As a result, a repeat failure occurred. Specifically, on October 23, 2000, while PVNGS, Unit 2, was in mode 6, the licensee identified that Unit 2, Train B, high pressure safety injection pump discharge Check Valve 2PSIBV405 had excessive back-leakage. This violation is in the licensee’s corrective action program as CRDR 2332280.

The issue was determined to be of very low safety significance because the core would have been adequately cooled, during the most severe accidents.

6. NCV 50-528/01-03-06

Technical Specification 5.7.1.b states, in part, that any individual or group of individuals permitted to enter a high radiation area shall be provided with a radiation monitoring device that continuously integrates the radiation dose rate in an area. On April 4, 2001, the licensee identified that between March 27 and April 4, 2001, 37 individuals using four different nonfunctioning electronic dosimeters entered high radiation areas. The cause of the electronic dosimeter problem was a vendor related firmware problem. The failure to wear a radiation monitoring device that continuously integrated the radiation dose rate in a high radiation area is a violation of Technical Specification 5.7.1. These events are described in the licensee's corrective action program, reference CRDR 2376601.

The safety significance of this finding was determined to be very low (Green) by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or unintended dose as a result of these nonfunctioning dosimeters.

ATTACHMENT

Supplemental Information

KEY POINTS OF CONTACTS

Licensee
S. Bauer, Section Leader, Regulatory Affairs
B. Blackmore, System Engineer, Safety Injection System
P. Borchert, Unit 2 Department Leader, Operations
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D. Mauldin, Vice President, Engineering and Support
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M. Sontag, Section Leader, Nuclear Assurance
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LIST OF ITEMS OPENED AND CLOSED

Opened

50-528/01-03-01	NCV	Isolation of reactor level instrumentation during partial drain conditions (Section 4OA7.1)
50-530/01-03-02	NCV	Failure to discontinue a reactor startup when required by procedure (Section 4OA7.2)
50-529/01-03-03	NCV	Auxiliary feedwater pump inoperable due to improper operation of a steam trap (Section 4OA7.3)

50-529/01-03-04	NCV	Failure to promptly identify HPSI valve misalignment on condition report/disposition request (Section 4OA7.4)
50-529/01-03-05	NCV	Failure to prevent recurrence of HPSI check valve failure, a significant condition adverse to quality (Section 4OA7.5)
50-528/01-03-06	NCV	Failure to wear a radiation monitoring device that continuously integrated the radiation dose rate in a high radiation area (Section 4OA7.6)

<u>Closed</u>

50-528/01-03-01	NCV	Isolation of reactor level instrumentation during partial drain conditions (Section 4OA7.1)
50-530/01-03-02	NCV	Failure to discontinue a reactor startup when required by procedure (Section 4OA7.2)
50-529/01-03-03	NCV	Auxiliary feedwater pump inoperable due to improper operation of a steam trap (Section 4OA7.3)
50-529/01-03-04	NCV	Failure to promptly identify HPSI valve misalignment on condition report/disposition request (Section 4OA7.4)
50-529/01-03-05	NCV	Failure to prevent recurrence of HPSI check valve failure, a significant condition adverse to quality (Section 4OA7.5)
50-528/01-03-06	NCV	Failure to wear a radiation monitoring device that continuously integrated the radiation dose rate in a high radiation area (Section 4OA7.6)
50-528;529;530/ 2000-004-00	LER	Technical Specification violation due to deficient test procedure for refueling purge valves (Section 4OA3.1)
50-528/2000-001-00	LER	Reactor Coolant System (RCS) pressure boundary leakage caused by degraded Alloy 600 instrument nozzle (Section 4OA3.2)
50-529/2000-005-00;01	LER	Safety injection discharge back-leakage causes degraded safety injection flow (Section 4OA3.3)

LIST OF DOCUMENTS REVIEWED

PROCEDURES

PROCEDURE	TITLE	REVISION
31MT-9PW02	Installation and Removal of Temporary Cooling Towers to NC Heat Exchanger for PW System Outage	0
40AO-9ZZ21	Acts of Nature	10
40EP-9EO07	Loss of Offsite Power/Loss of Forced Circulation	6
40OP-9DG01	Emergency Diesel Generator A	16
40OP-9PC01	Fuel Pool Cooling	2
40OP-9SI01	Shutdown Cooling Initiation	22
40OP-9SI02	Recovery from Shutdown Cooling to Normal Operating Lineup	36
40OP-9ZZ11	Mode Change Checklist	43
40ST-9DG01	Diesel Generator A Test	14
40ST-9SI04	Containment Spray Valve Verification	1
40ST-9SI07	High Pressure Safety Injection System Alignment Verification	5
40TD-9PC01	Fuel Pool Cooling and Cleanup	2
42OP-9ZZ04	Plant Startup Mode 2 to Mode 1	25
70DP-0MR01	Maintenance Rule	5
71DP-OEM01	Risk Management Program Expert Panel	5
72OP-9RX01	Calculation of Estimated Critical Condition	8
72OP-9RX02	Determination of Anticipated Critical Position	3
73TI-9AF04	Feeding SG's Using AFB-P01 to Verify AFA-FI-40A and Associated Equipment	0
90DP-OIP10	Condition Reporting	11

Drawings

01-M-PCP-001, "Fuel Pool Cooling and Cleanup System," Revision 22
01-M-SIP-001/002, “Safety Injection and Shutdown Cooling System,” Revision 25
13-M-DGP-001, “Emergency Diesel Generators”
13-M-DFP-001, “Fuel Oil Transfer and Storage System”

Condition Report/Disposition Requests

* Includes Cause Determination

2332280*	2382744	96Q067	117493
2335098	2377394	98Q311	117937
2345083	2379228	2399032	290131
2356232*	2345330	2380869	2308926
2379175	2382916	34879*	
2382028	920111	45252	
2387300			

MAINTENANCE RULE PROGRAM EXPERT PANEL MEETING MINUTES

January 3, 1997	April 16, 1998	August 25, 1998
March 26, 1998	June 11, 1998	February 8, 2001

ENGINEERING REPORTS

DESCRIPTION	DATE
Maintenance Rule Self-Assessment Report for the period of March 1998 - August 1999	January 14, 2000
Maintenance Rule Self-Assessment Report for the period of September 1999 - December 2000	April 17, 2001

MISCELLANEOUS

NUMBER	DESCRIPTION	REVISION
Updated Final Safety Analysis Report		10
Section 3.2	Classification of Structures, Systems, and Components	
Section 3.3	Wind and Tornado Loading	
Section 3.5	Missile Protection	
Section 7.4	Flood Protection	
Calculation 13-NC-SP-206	Ultimate Heat Sink Design Reverification	21

LIST OF ACRONYMS

ASI	axial shape index
CFR	Code of Federal Regulations
CRDR	condition report/disposition request
LER	licensee event report
NCV	noncited violation
RCS	reactor coolant system
SSC	structures, systems, and components
URI	unresolved item