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

August 27, 2004

Greg R. Overbeck, Senior Vice
President, Nuclear
Arizona Public Service Company
P. O. Box 52034
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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - NRC SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY INSPECTION REPORT 05000528/2004-007, 05000529/2004-007, AND 05000530/2004-007

Dear Mr. Overbeck:

On July 16, 2004, the Nuclear Regulatory Commission (NRC) completed an inspection at your Palo Verde Nuclear Generating Station, Units 1, 2, and 3. The enclosed report documents the inspection findings, which were discussed on July 16, 2004, with Mr. David Mauldin, Vice President, Engineering, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified one finding. This finding does not present an immediate safety concern because the licensee had performed a probabilistic risk assessment indicating that the probability of a tornado-generated missile affecting the auxiliary feedwater minimum flow recirculation line was minimal. The NRC has also determined that a violation was associated with this finding. This violation was determined to be a noncited violation of very low safety significance. The violation is described in the subject inspection report. If you contest the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; 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 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/

Jeff Clark, P.E., Chief
Engineering Branch
Division of Reactor Safety

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

Enclosure:
NRC Inspection Report 05000528; 529; 530/2004007
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-528, 50-529, 50-530
Licenses:	NPF-41, NPF-51, NPF-74
Report No:	05000528/2004-007; 05000529/2004-007; 05000530/2004-007
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station, Units 1, 2, and 3
Location:	5951 S. Wintersburg Tonopah, Arizona
Dates:	June 7 through July 16, 2004
Team Lead:	Claude E. Johnson, Senior Reactor Inspector, Engineering Branch
Inspectors:	J. P. Adams, Reactor Inspector, Engineering Branch J. M. Mateychick, Reactor Inspector, Engineering Branch T. Farnholtz, Senior Project Engineer, DRP B. W. Henderson, Reactor Inspector, Engineering Branch
Accompanied By:	G. Skinner, Electrical Contractor, Beckman Inc.
Approved By:	Jeff Clark, P.E., Chief Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000528/2004007; 05000529/2004007; 05000530/2004007; 6/07/04 through 7/16/04 Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Safety System Design and Performance Capability Inspection and Evaluation of Changes, Tests, or Experiments.

The report covered a 2-week onsite inspection by five regional inspectors and one contractor. During the onsite inspection week of June 7-14, 2004, all 3 units tripped on June 14 due to a loss-of-offsite power. The team departed on June 15, 2004, due to an Augmented Inspection Team sent to the site because of the loss-of-offsite power. The team resumed the second onsite week of the inspection July 14, 2004.

The inspection identified one Green noncited violation. The significance of most findings is indicated by its color (Green, White, Yellow, 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 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.

<u>NRC-Identified</u>

Cornerstone: Mitigating System

<u>GREEN</u>. The team identified an noncited violation for the failure to comply with 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee failed to correctly translate design information into the as-built configuration of the auxiliary feedwater system, in that, 28 feet of exposed auxiliary feedwater minimum flow recirculation line was not protected from a tornado-generated missile for both trains as described in Design Basis Manual, Table 2-1 and Section 10.4.9.1, "Design Basis," of the Final Safety Analysis Report. This issue was entered into the licensee's corrective action program as Condition Report/Deficiency Request 2721947.

In accordance with NRC Inspection Manual 0612, Appendix B, "Issue Screening," this finding is greater than minor because it is associated with the design control attribute of the mitigating systems cornerstone, and affected the cornerstone objective to ensure the capability of systems to respond to initiating events. The inspectors evaluated the issue using the Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones provided in Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The finding was determined to be of very low safety significance because: the finding did not represent an actual loss-of-safety function and because the analyst determined that the system would continue to meet its risk-significant function following a postulated tornado initiating event. (Section 1R21.4)

1. REACTOR SAFETY

Introduction

The NRC conducted an inspection to verify that the licensee adequately preserved the facility safety system design and performance capability and that the licensee preserved the initial design requirements in subsequent modifications of the systems selected for review. The scope of the review also included any necessary nonsafety-related structures, systems, and components that provided functions to support safety functions. This inspection also reviewed the licensee's programs and methods for monitoring the capability of the selected systems to perform the current design basis functions. This inspection verified aspects of the mitigating systems, and barrier integrity cornerstones.

The licensee based the probabilistic risk assessment model for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, on the capability of the as-built safety systems to perform their intended safety functions successfully. The team determined the area and scope of the inspection by reviewing the licensee's probabilistic risk analysis models to identify the most risk significant systems, structures, and components. The team established this according to their ranking and potential contribution to dominant accident sequences and/or initiators. The team also used a deterministic approach in the selection process by considering recent inspection history, resident inspector feedback, recent problem area history, and modifications developed and implemented.

The team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that the licensee used for the selected safety systems and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria used by the NRC inspectors included NRC regulations, the technical specifications, applicable sections of the Updated Final Safety Analysis Report, applicable industry codes and standards, and industry initiatives implemented by the licensee's programs.

The minimum sample size for this procedure is one or two risk-significant systems for mitigating an accident or maintaining barrier integrity. The team completed the required sample size by reviewing the auxiliary feedwater system, reactor protection system, and safety-related portions of the instrument air system. The primary review prompted parallel review and examination of support systems, such as, electrical power, instrumentation, and related structures and components.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The team reviewed eight licensee-performed 10 CFR 50.59 evaluations to verify that the licensee had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. These evaluations had been performed since the last NRC inspection of 10 CFR 50.59 activities.

The team reviewed an additional 14 licensee-performed 10 CFR 50.59 screenings, in which the licensee determined that evaluations were not required, to ensure that the licensee's exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59.

The team reviewed and evaluated the most recent licensee 10 CFR 50.59 program self-assessment and nine corrective action documents written since the last NRC 10 CFR 50.59 inspection to determine whether the licensee conducted sufficient in-depth analysis of their program to allow for the identification and subsequent resolution of problems or deficiencies.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Requirements

a. Inspection Scope

The team inspected the following attributes of the auxiliary feedwater system, instrument air system and the reactor protection system: (1) process medium (water, steam, and air), (2) energy sources, (3) control systems, and (4) equipment protection. The team examined the procedural instructions to verify that instructions are consistent with actions required to meet, prevent, and/or mitigate design basis accidents. The team also considered requirements and commitments identified in the Updated Final Safety Analysis Report, technical specifications, design basis documents, and plant drawings. In conjunction with the primary review of the safety-related systems, a parallel review and examination of support systems, such as, the non-safety auxiliary feedwater pump, safety-related accumulators when instrument air is unavailable, and related structures and components was also conducted.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The team reviewed the periodic testing procedures for the auxiliary feedwater, reactor protection, and instrument air systems to verify that the capabilities of the systems were verified periodically. The team also reviewed the systems' operations by conducting system walkdowns; reviewing normal, abnormal, and emergency operating procedures; and reviewing the Updated Final Safety Analysis Reports, technical specifications, design calculations, drawings, and procedures.

b. Findings

No findings of significance were identified.

.3 System Walkdowns

a. Inspection Scope

The team performed walkdowns of the accessible portions of the selected systems and support systems. The team focused on the installation and configuration of switchgear, motor control centers, manual transfer switches, field cabling, raceways, piping, components, and instruments. During the walkdowns, the team assessed:

- The placement of protective barriers and systems,
- The susceptibility to flooding, fire, or environmental conditions,
- The physical separation of trains and the provisions for seismic concerns,
- Accessibility and lighting for any required operator action,
- The material conditions and preservation of systems and equipment, and
- The conformance of the currently-installed system configurations to the design and licensing bases.

b. Findings

No findings of significance were identified.

.4 Design Review

a. Inspection Scope

The team reviewed the current as-built instrument and control, electrical, and mechanical design of the selected systems and support systems. These reviews included an examination of design assumptions, calculations, environmental qualifications, required system thermal-hydraulic performance, electrical power system performance, control logic, and instrument setpoints and uncertainties. The team assessed the adequacy of calculations, analyses, test procedures, and operating procedures that licensee personnel used during normal and accident conditions.

b. Findings

Introduction. The team identified an violation for failure to correctly translate design information into the as-built configuration, specifically, the auxiliary feedwater minimum flow recirculation lines.

<u>Description</u>. During a system walkdown, the team identified a section of the auxiliary feedwater minimum flow recirculation line, 28 feet in length, associated with both essential auxiliary feedwater trains, that extends outside the concrete protective bunker and is not protected from natural phenomena, such as tornado-generated missiles. These lines are used to ensure that the auxiliary feedwater pumps, when started as the result of an accident, will not pump against an isolated line and potentially be damaged. The concern associated with the lack of missile protection is that if damaged by a tornado-generated missile, the auxiliary feedwater system could potentially pump up to 1/4 of its flow through the minimum flow recirculation line onto the ground, thereby, reducing the total volume of water in the condensate storage tank that would be available for decay heat removal.

The team's review of Design Basis Manual Table 2-1 and Section 10.4.9.1, "Design Basis," of the Final Safety Analysis Report indicated that the auxiliary feedwater system is required to be protected from tornado-generated missiles. The system walkdown revealed that the as-built configuration did not meet the design basis documents. The team informed the licensee of the finding. The licensee presented the team with copies of an Safety Evaluation Report, Supplement 1, and a probabilistic calculation (13-NC-CT-200). The Safety Evaluation Report was submitted to demonstrate compliance with General Design Criteria 2, which requires tornado missile protection for the ultimate heat sink (spray ponds) and associated safety-related piping. The probabilistic calculation was given to the team to provide proof that the unavailability of non-safety portion of the condensate storage tank and associated piping, including the exposed portions of the auxiliary feedwater recirculation piping if struck by a tornado-generated missile would be minimal.

Although the team acknowledged the Safety Evaluation Report and the calculation and determined it to be reasonable, the team informed the licensee that the NRC staff's approval of the Safety Evaluation Report also stated that acceptance of the above plant-specific tornado missile assessment is not a generic acceptance of this method. In other words, it is acceptable for the ultimate heat sink only, and cannot be used for other systems and applications. The team also determined that the calculation did not specifically address the effects of a loss-of-auxiliary feedwater system if damage occurred to the recirculation lines, which could potentially affect the reserved condensate inventory and the auxiliary feedwater pumps. Specifically, if the minimum flow recirculation line were damaged and the auxiliary feedwater pumps continued to pump up to 1/4 of the total flow through the break the volume of water in the condensate storage tank might not be sufficient from the auxiliary feedwater safety function. The team informed the licensee that the probabilistic assessment for the auxiliary feedwater recirculation line should have been submitted to the NRC staff for approval.

The team did not consider that this represented an immediate safety concern because the licensee had performed a probabilistic calculation that showed the probability of a tornado-generated missile damaging piping lines into the condensate storage tank is very low.

The team concluded that the licensee had violated 10 CFR Part 50, Appendix B, Criterion III, by not constructing the plant in accordance with the design basis, as described in the Final Safety Analysis Report, in that, the Final Safety Analysis Report states that the auxiliary feedwater system will be protected against adverse environmental conditions. However, the team determined, during a walkdown of the auxiliary feedwater system, the licensee had failed to protect the auxiliary feedwater minimum flow recirculation lines from tornado-generated missiles. The cause of the violation is that the licensee failed to adequately control the design process, in that, they changed the plant from its original design without making a concomitant change to the design basis documents.

Analysis. The team determined that the licensee's failure to accurately translate the design basis documents into the as-built configuration is a performance deficiency because the licensee is expected to meet the requirements of 10 CFR Part 50, Appendix B, Criterion III. This finding is greater than minor because it affected the mitigating systems cornerstone, specifically, because both trains of auxiliary feedwater minimum flow recirculation lines were not protected against external factors, such as tornado-generated missiles as described in Design Basis Manual, Table 2-1, and Section 10.4.9.1, "Design Basis," of the Final Safety Analysis Report.

A regional senior reactor analyst evaluated the capability of the system to perform its risk-significant function, given that a tornado causes both a loss-of-offsite power and a failure of the pump minimum flow recirculation lines. The analyst determined, through rigorous calculation of the auxiliary feedwater flow requirements, that in most cases, the auxiliary feedwater system would continue to function throughout the 24-hour probabilistic risk assessment mission time. The analyst determined that the condensate storage tank and the reactor makeup water tank contained a combined volume of water sufficient to cool the plant at hot standby for 24 hours.

However, during a loss-of-offsite power with an automatic auxiliary feedwater system initiation, these tanks did not contain a sufficient volume of water to meet a 24-hour mission without some outside intervention. The analyst determined that there were three independent success paths under this configuration: 1) Plant personnel would observe the large leakage in the yard, and inform plant operators who would isolate the leakage; 2) Licensed operators would observe the reduction in tank levels and proceed to cooling the plant and placing it in shutdown cooling, prior to depleting auxiliary feedwater water sources; 3) Technical Support Center personnel would determine a need to make up water to one of the two suction tanks prior to inventory depletion.

The analyst determined that the probability of failure for each of these success paths was statistically independent from the others because of the separation in time, location, methods, and personnel responding. Using the SPAR-H method for determining human error probabilities the analyst calculated the product of each of the three independent probabilities. The calculated probability of failure for the auxiliary feedwater system during this scenario was 2.4×10^{-7}.

The analyst concluded that the auxiliary feedwater system would continue to operate throughout it's risk-significant mission time for all scenarios. The water volumes maintained in the condensate storage tank and reactor makeup water tank were sufficient to respond to all but a small fraction of events. In the rare event that a tornado causes an unrecoverable loss-of-offsite power and ruptures the recirculation lines in the auxiliary feedwater system, and an automatic actuation of the auxiliary feedwater system is required, the analyst calculated that the probability that the licensee failed to ensure that sufficient water was made available was sufficiently small that it would not affect the risk function of the system.

Enforcement. The Final Safety Analysis Report states, in Section 10.4.9.1, "This system (AFW) shall be designed such that adverse environmental conditions such as tornados, floods, and earthquakes will not impair its safety function." Appendix B, Criterion III, in 10 CFR Part 50 states, in part, that ". . . [m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in paragraph 50.2 and as specified in the license application . . . are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled." Contrary to this Final Safety Analysis Report design criterion, the licensee did not protect part of the auxiliary feedwater system. Specifically, the minimum flow recirculation lines for both auxiliary feedwater trains are not protected from tornado-generated missiles, as required by design basis documents. Because of the very low safety significance of the finding and because the finding has been entered into the corrective action program as Condition Report/Deficiency Request 2721947, the team treated this as a noncited violation: NCV 05000528, 05000529, 530/2004007-01, Failure to correctly translate design information into the as-built configuration.

.5 Safety System Inspection and Testing

a. Inspection Scope

The team reviewed the program and procedures for testing and inspecting selected components for the selected systems and support systems. The review included the results of surveillance tests required by the technical specifications and selective review of inservice tests.

b. Findings

No findings of significance were identified.

4. Other Activities

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems associated with selected systems and support systems that were identified by licensee personnel in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The sample included open and closed condition reports for the past 3 years, which are listed in the attachment to this report. Inspection Procedure 71152, "Identification and Resolution of Problems," was used as guidance to perform this part of the inspection. Older condition reports that were identified while performing other areas of the inspection were also reviewed.

b. Findings

No findings of significance were identified.

4OA5 Other

a. Scope

As part of this inspection effort the team reviewed certain aspects of the reactor protection system, such as wiring diagrams, logic diagrams, general arrangement drawings, the Final Safety Analysis Report, reactor protection system calculations and surveillances, and reactor protection system corrective actions. The team reviewed motor-operated valve thrust calculations, and reviewed calculations to determine voltage at Generic Letter 89-10 motor-operated valves for use in thrust calculations. The calculations only modeled one motor-operated valve running at a time and did not consider bus loading due to several motor-operated valves starting or running at the same time, such as would occur during an actual event. The licensee response determined that other conservatisms in the calculation were sufficient to account for the voltage deficit resulting from incorrect modeling of running and starting loads, and reviewed the modification package for the core protection calculator for Unit 2. The team also performed a visual inspection of the unmodified Unit 3 reactor protection system, as well as, the modified Unit 2 reactor protection system with the new core protection calculator.

b. Findings

There were no findings in the areas above. However, during review of the repair and fabrication of reactor protection system circuit cards, the team could not complete the inspection during this period due to its complexity. The team noted that repair and fabrication of circuit cards were performed onsite by the licensee. The team's brief review preliminary determined that quality assurance of the reactor protection system circuit cards was suspect. It was decided to continue this inspection effort by the Resident Inspectors in the next quarterly report.

4OA6 Management Meetings

Exit Meeting Summary

The inspection findings were acknowledged during an exit meeting presented by the team leader on July 16, 2004, to Mr. David Mauldin, Vice President, Engineering, and other members of licensee management staff. The team leader confirmed that proprietary information, while reviewed, had not been retained by the team.

ATTACHMENT

KEY POINTS OF CONTACT

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Others

R. Henry, Site Representative, Salt River Project

NRC personnel

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N. Salgado, Senior Resident Inspector, Palo Verde Nuclear Station

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000528, 529, 530/2004007-01	NCV	Failure to correctly the design basis documents into the as-built configuration (Section 1R21.4).

DOCUMENTATION REVIEWED

Calculations

NUMBER	DESCRIPTION	REVISION
02-NC-SG-200	ADV Reliability Analysis	1
13-JC-AF-0002	Auxiliary Feedwater Flow Loops (J-AFA-F-0040A/B & J-AFB-F-0041A/B)Total Loop Uncertainty Calculation	7
13-JC-RC-0207	Pressurizer Pressure (PPS High) Instrument (RCx-P-101x; x=A,B,C,D) Uncertainty and Setpoint Calculation	10
13-JC-SG-0201	Atmospheric Dump Valve (ADV) Nitrogen Accumulator Tank Pressure Loop (13-J-SGA-P-0308/0315 & 13-J-SGB-P-301/321) Setpoint and Uncertainty Calculation	3
13-JC-SG-0205	Uncertainty Calculation for MSIV/FWIV Air Reservoir Pressure Channels	4
13-JC-ZZ-201	MOV Thrust, Torque and Actuator Sizing Calculation	13
13-MC-AF-0309	AF Hydraulic Calculation for Q-Trains	7
13-MC-AF-401	AFW System - MOV Maximum Differential Pressure	3
13-MC-AF-800	Auxiliary Feedwater ESF Function Response Times	6
13-MC-CT-0205	Condensate Storage Tank	3
13-MC-CT-030	CST Minimum Level Setpoint	3
13-MC-SG-0314	Nitrogen Tank Pressure Requirements for ADVs	5
13-MC-SG-0316	MSIV and FWIV - Verification of Volume and Pressure Drop of Air Reservoir	7
13-MC-SG-0318	Pressure/Temperature Rating of N2 Back UP System for ADV's	1
13-MC-SG-402	Feedwater Isolation Valves and Control Valves Actuator Gas Volume Verification	2 with EDC # 98-000423
13-MC-SG-405	ADV Nitrogen Tank Temperature Adjusted Pressures	2
13-MC-SG-811	Maximum Differential Pressure Across MOV's SG-134/138 and AF-54	2
13-MC-ZA-807	MSSS Building Flooding at Elevation 100'	3
13-MC-ZA-808	MSSS Flooding at Elevation 81'	3

NUMBER	DESCRIPTION	REVISION
13-NC-CT-200	CST Tornado Damage Analysis - Availability of non-safety portion of CST inventory	0
13-NC-SG-001	MSIV 50.59 Reliability Evaluation	1

Drawings

NUMBER	DESCRIPTION	REVISION
01-E-PBA-002	Single Line Diagram 4.16t KV Class 1E Power System Switchgear 1E-PBB-S04	10
01-E-PBB-002	Single Line Diagram 4.16 KV Class 1E Power System Switchgear 1E-PBB-S04	10
01-E-PHA-004	Single Line Diagram 480 V Class 1E Power System Motor Control Center 1E-PHB-M34	18
01-E-PHA-008	Single Line Diagram 480V Class 1E Power System Motor Control Center 1E-PHB-M38	14
01-E-PKA-002	Single Line Diagram 125V DC Class 1E Power System DC Control Center 1E-PKA-M41	15
01-E-PKA-004	Single Line Diagram 125V DC Class 1E Power System DC Control Center 1E-PKA-M43	7
01-J-SCE-062	Instrument Loop Wiring Diagram Reactor Coolant System	6
01-J-SCE-064 Sh 1	Instrument Loop Wiring Diagram Main Steam System	6
02-E-SBF-001, SH 1	Control Wiring Diagram Reactor Protection System Reactor Trip Breaker Channel A	4
02-E-SBF-001, SH 2	Control Wiring Diagram Reactor Protection System Reactor Trip Breaker Channel B	4
02-E-SBF-001, SH 3	Control Wiring Diagram Reactor Protection System Reactor Trip Breaker Channel A	4
02-E-SBF-001, SH 4	Control Wiring Diagram Reactor Protection System Reactor Trip Breaker Channel B	4
02-E-SBF-003, SH 1	Control Wiring Diagram Reactor Protection System 120V AC Supply	2
02-E-SBF-003, SH 2	Control Wiring Diagram Reactor Protection System 120V AC Supply	2

NUMBER	DESCRIPTION	REVISION
02-E-SBF-003, SH 3	Control Wiring Diagram Reactor Protection System 120V AC Supply	2
02-E-SBF-003, SH 4	Control Wiring Diagram Reactor Protection System 120V AC Supply	2
02-E-SBF-003, SH 5	Control Wiring Diagram Reactor Protection System 120V AC Supply	2
02-E-SBF-007, SH 1	Control Wiring Diagram Plant Protection System Channel C-Part 1	5
02-E-SBF-007, SH 2	Control Wiring Diagram Plant Protection System Channel C-Part 2	5
02-E-SBF-007, SH 3	Control Wiring Diagram Plant Protection System Channel C-Part 5	5
02-E-SBF-007, SH 4	Control Wiring Diagram Plant Protection System Channel C-Part 6	6
02-E-SBF-007, SH 5	Control Wiring Diagram Plant Protection System Channel C-Part 7	5
02-E-SBF-007, SH 6	Control Wiring Diagram Plant Protection System Channel C-Part 8	5
02-E-SBF-007, SH 7	Control Wiring Diagram Plant Protection System Channel C-Part 9	5
02-E-SBF-007, SH 8	Control Wiring Diagram Plant Protection System Channel C-Part 10	5
02-E-SBF-007, SH 9	Control Wiring Diagram Plant Protection System Channel C-Part 11	5
02-E-SBF-007, SH 10	Control Wiring Diagram Plant Protection System Channel C-Part 12	5
02-E-SBF-007, SH 11	Control Wiring Diagram Plant Protection System Channel 2C-Part 13	5
02-M-AFP-001	P& I Diagram Auxiliary-Feedwater System	24
02-M-CTP-001	P& I Diagram Condensate Storage and Transfer System	18
02-M-SGP-001	P& I Diagram Main Steam System	53
02-M-SGP-002	P& I Diagram Main Steam System	35
03-M-AFP-001	P&ID Auxiliary-Feedwater System	21

NUMBER	DESCRIPTION	REVISION
03-M-CTP-001	P&ID Condensate Storage and Transfer system	15
03-M-SGP-001	P&ID Main Steam System	42
03-M-SGP-002	P&ID Main Steam System	30
13-P-ZYA-013	Condensate Water Storage Tank Piping Plan	25
13-P-ZYA-014	Condensate Water Storage Tank Piping Section	17
6474-33043 Sh 1	Plant Protection System General Information	0
6474-33043 Sh 2	Plant Protection System General Information	0
6474-33043 Sh 2	Plant Protection System Assembly Cabinet	0
6474-33043 Sh 3	Plant Protection System General Information	0
6474-33043 Sh 4	Plant Protection System General Information	0
6474-33100 Sh 1	Plant Protection System Assembly Cabinet	B
6474-33100 Sh 5	Plant Protection System General Information	0
D-SYS80-411-534	PPS Initiation Circuit Schematic	2
E-SYS80-411-501 Sh 1	Plant Protection System Simplified Functional Diagram	2
E-SYS80-411-501 Sh 2	Plant Protection System Simplified Functional Diagram	2
E-SYS80-411-501 Sh 3	Plant Protection System Simplified Functional Diagram	2
E-SYS80-411-503	Plant Protection System Logic Diagram	2
E-SYS80-413-130	Reactor Trip Switchgear System (RTSS) Arrangement &Control Wiring Diagram	4

<u>Procedures</u>

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36ST-9SA01	ESFAS Train A Subgroup Relay Functional Test	29
36ST-9SB02	PPS Bistable Trip Units Functional Test	28
36ST-9SB58	PPS Transmitter Input Calibration for Parameters 5,6,11,14, and 15	6

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36ST-9SB59	PPS Input Loop Calibration for Channel A Parameters 5,6,11,12, 14 and 15	8
40AL-9DG01	Diesel Generator A Alarm Panel Responses	16
40AO-9ZZ06	Loss of Instrument Air	14
40EP-9E003	Loss of Coolant Accident	14
40EP-9E004	Steam Generator Tube Rupture	15
40EP-9EO10	Standard Appendices	32
40ST-9ZZM1	Operations Mode 1 Surveillance Logs	29
41AL-1RK2A	Panel B02A Alarm Responses	43
42AL-2RK6A	Annunciator Window Index - CST TRBL	27
73ST-9AF01	AFN-P01-Inservice Test	7
73ST-9AF02	AFA-P01 - Inservice Test	30
73ST-9AF03	AFB-P01 - Inservice Test	14
73ST-9DG01	Class IE Diesel Generator and Integrated Test - Train A	8
73ST-9SG01	MSIVs - Inservice Test	19
73ST-9SG05	ADV Nitrogen Accumulator Drop Test	16
73ST-9XI05	AF and CT Valves - Inservice Test	21
73ST-9XI16	Economizer FWIVs - Inservice Test	18
73ST-9XI20	Atmospheric Dump Valves (ADV) - Inservice Test	16
73ST-9XI36	AFA-P01 Steam Supply Check Valves - Inservice Test	4
80DP-0DC01	Reverse Engineering and Manufacturing Process	2
93DP-0LC07	10 CFR 50.59 and 72.48 Screenings and Evaluations	7

Miscellaneous Documents

Atmospheric Dump Valve Engineering Analysis, dated March/April, 1989

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Design Modification Work Order # 218265

EER# 89-SG-238, “Backup N2 Accumulators for the Atmospheric Dump Valves”

EER# 90-SG-002, “ADV Nitrogen Check Valves”

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Nuclear Administrative and Technical Manual 73DP-9X101, “Pump and Valve Inservice Testing Program - Component Tables:, Revision 14

System Training Manual Vol. 21; Auxiliary Feedwater System, Rev. 5

Work Order 2492915

Work Order 2556988

Work Order 2590651

Design Base Documents

PVNGS Design Basis Manual, Auxiliary Feedwater System, Revision 014

PVNGS Design Basis Manual, Instrument Air System, Revision 008

PVNGS Design Basis Manual, Main Steam System, Revision 018

Technical Specification (through Amendment 150) Section 3.3.5 Engineered Safety Systems Actuation System (ESSAS) Instrumentation

Technical specification (through Amendment 150) Section 3.7.5 Auxiliary Feedwater System

Technical specification (through Amendment 150) Section 3.7.6 Condensate Storage Tank

Technical Specification Bases B 3.7.6 Condensate Storage Tank (CST) Rev 27

UFSAR Sections

Revision 12 - List B, December 2003
6.3.3.3, “Small Break Analysis”
7.0, “Instrumentation and Controls”
9.3.1.1.2.1, “Instrument Air System”
10.3, “Main Steam Supply System”
10.4.9, “Auxiliary Feedwater System”
15.2, “Decrease in Heat Removal by the Secondary System”
15.3, “Decrease in Reactor Coolant Inventory”
Appendix 15A, “Responses to NRC Requests for Information”

Amendment Changes

126, Unit 1, 2, and 3, Reactor Protection System Instrumentation: Revised TS 3.3.1, “RPS Instrumentation - Operating; Change the allowable values for two of the trip setpoints. Dated July 6, 2000

133, Unit 1, 2, and 3, Change the Minimum Departure Of the Nucleate Boiling Ratio, dated March 28, 2001

134, Unit 1, 2, and 3, Amendments on 7-day completion time for turbine-driven auxiliary feedwater system, dated March 29, 2001

135, Unit 1, 2, and 3, Amendments on response time testing for ESF and Reactor Pressure System, dated April 19. 2001

150, Unit 1, 2, and 3, Amendment to CorePC system upgrade, dated October 24, 2003

APS Letter 102-04310-WEI/SAB/RKR, Response to NRC Request for Additional Information Regarding Proposed Amendment to Technical Specifications (TS) 3.8.1, AC Sources-Operating and 3.3.7, Diesel Generator (DG)-Loass of Voltage Start (LOVS), dated July 16, 1999

DSG-IC-0205, Design Guide for Instrument Uncertainty and Setpoint Determination, Revision 9

NRC Letter S.A. Richards NRR, to P. Richardson Westinghouse, Acceptance for Referencing of Topical Report CENPN-396-P Rev. 01, ?Common Qualified Platform? and Appendicies 1, 2, 3, and 4, Rev. 01 (TAC No. MA1677), dated August 11, 2000

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Westinghouse Report 00000-ICE-37738, EMI Qualification Test Report of Supplemental Testing for Common Q Applications, Revision 00

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DMWO 223535, Replace CPC/CEAC system due to system obsolescence and spare parts vailability issues, date November 4, 1999

10 CFR 50.59 Screenings

S-02-0092	S-02-0235	S-03-0167	S-03-0212	S-03-0289
S-02-0101	S-02-0266	S-03-0184	S-03-0221	S-03-0335
S-02-0208	S-02-0270	S-03-0211	S-03-0269	

Condition Reports/ Deficiency Request (CRDRs)

115731	2303258	2357581	2442997	2595572	2641052
115842	2304210	2362432	2450703	2597124	2646484
115978	2304413	2364040	2456094	2602675	2647844
116857	2304747	2381684	2467532	2602686	2650307
117294	2305041	2399032	2467852	2604274	2650400
117494	2317329	2407667	2477693	2622513	2659922
117862	2318274	2409548	2508869	2625374	2667663
117868	2322502	2418786	2540877	2630504	2682317
118160	2339523	2425664	2540920	2632300	2714983
118352	2339678	2427587	2573398	2635258	2715252
118492	2345062	2430089	2575739	2636177	2715285
266983	2348260	2439147	2583750	2636488	
703945	2348844	2442546			

10 CFR 50.59 Evaulations

E-02-0006	E-02-0026	E-02-0029
E-02-0011	E-02-0028	E-03-0002
E-02-0020		