ARIZONA PUBLIC SERVICE COMPANY

PALO VERDE NUCLEAR GENERATING STATION

CONTROL ROOM SIMULATOR

TRAINING SUITES A & B

FOUR YEAR CERTIFICATION REPORT

TABLE OF CONTENTS

1.0 Purpose

The Palo Verde Simulators are certified to meet the requirements of 10CFR55.45 as a valid tool for conducting the simulator portion of NRC license examinations, and for meeting the licensed operator initial, upgrade, and requalification training objectives of Palo Verde. In accordance with 10CFR55.45, every four years on the anniversary of the certification application, a report must be submitted to the Commission.

The purpose of this report is to satisfy this reporting requirement for the Palo Verde Control Room Simulators "A" and "B".

2.0 References

A. Title 10, Code of Federal Regulations, Part 55, "Operator Licenses", Subpart E, Section 45.

B. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.149 "Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations", April 1987.

C. ANSI/ANS 3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training."

3.0 Simulator and Plant Description

A. Reference Plant General Information

Docket No:	50-528
Operator:	Arizona Public Service Company
Location:	Palo Verde Nuclear Generating Station, Wintersburg, Arizona
Manufacturer:	Combustion Engineering (NSSS)
	Bechtel (AE)
MW Output:	1270 MWe
NSSS:	2-Loop PWR
Commercial Operation:	1985 (Unit 1)

B. Simulator General Information

Operator:	Arizona Public Service Company
Location:	Palo Verde Nuclear Generating Station, Wintersburg, Arizona
Manufacturer:	S3 Technologies Incorporated
Model Development:	S3 Technologies Incorporated
Reference Plant	Palo Verde Unit 1
Date Available for Training	November 1993
Simulator "A" Initial Certification Submittal:	December 29, 1993
Certification Submittal Update to include Simulators "A" and "B":	August 30, 1994

C. Computer Configuration

There were no major modifications to the simulation facility in the last four years. The DEC/A was added in 1996 for stimulating fire console response in simulator suite "A".

Main Simulation Computer:	Silicon Graphics Challenge with CPI I/O interface; SCRAMNET Fiber Optic link to Encore 67 running the Plant Monitoring System; ERFDADS link via Ethernet.
Stimulation Computers:	DEC/PDP 11/34 (1)
	SUN Ultra (2)
	DEC/A, with no link (3)
Instructor Interface:	SGI Indigo X-Terminals (4)
Ambient Sound Generation:	80386 with Sound Generation Software (5)

Notes

(1) Simulated DEC Radiation Monitoring System receiving process data from SGI Challenge.

(2) Used for Emergency Response Facility Data Acquisition and Display System (ERFDADS), was not included on original certification submittal but was present as a SUN Spark station. Upgraded to SUN Ultra in 1998.

(3) Used for signal generation to fire console as a stand-alone unit, Simulator Suite "A" only. Added to simulator configuration in 1996.

(4) There are a total of two instructor station SGI Indigo X-Terminals.

(5) Ambient sounds are generated for shifts in ventilation and other events in Simulator Suite "A" only.

D. Exception to ANSI/ANS 3.5 - 1985

These exceptions and justifications are the same as the certification submittal for Simulators "A" and "B", dated August 30, 1994. They primarily deal with items related to Boiling Water Reactors, Power Operated Relief Valves and sustained critical operations with less than full reactor coolant flow. They are as follows:

1) All items in ANSI/ANS-3.5-1985 specific to Boiling Water Reactor (BWR) Plants.

   An exception for this is justified because Palo Verde is a Pressurizer Water Reactor. No BWR training is provided to the Palo Verde Operators.

2) Malfunctions affecting the operation of reactor coolant system Power Operated Relief Valves (PORVs).

   Palo Verde does not have PORVs installed in the reactor coolant system. No training on reactor coolant system PORVs is provided to the Palo Verde Operators.

3) Startup, shutdown and power operations with less than full reactor coolant flow.

   Palo Verde's reactor protection system does not permit power operation with less than full reactor coolant flow. Palo Verde is not analyzed for startup, shutdown or power operations with less than full reactor coolant flow. No training is

provided to the Palo Verde Operators on startup, shutdown or power operations with less than full reactor coolant flow (ANSI/ANS-3.5-1985, Section 3.1.1(7)).

4.0 Requirements

A. 10CFR55.45 requires the following be contained within this report:

1) Subpart E, Section 55.45 (b)(5)(ii) - Identification of any uncorrected performance test failures and a schedule for correction of such performance failures.

2) Subpart E, Section 55.45 (b)(5)(vi) - A description of performance testing completed for the simulation facility.

3) Subpart E, Section 55.45 (b)(5)(vi) - A description of the performance tests, if different, to be conducted on the simulation facility during the subsequent four year period.

4) Subpart E, Section 55.45 (b)(5)(vi) - A schedule for the conduct of approximately 25 percent of the performance tests per year for the subsequent four years.

5.0 Palo Verde Compliance with Requirements

A. Uncorrected performance test failures at Palo Verde and schedule for correction

1) There are no uncorrected performance test failures at Palo Verde

B. Description of performance testing performed in the last four year period.

ANS/ANSI 3.5-1985 Appendix A, section A.3, describes certification testing requirements. Four separate categories are identified and tested as follows:

1) Computer Real Time Test

This test was performed annually.

2) Steady State Tests

Steady State Testing is composed of three separate activities performed annually:

a) Steady state performance testing is performed to evaluate and benchmark the degree of agreement between the simulator and the Palo Verde Nuclear Generating Station Unit 1 for mass and energy balance parameters at three steady state power levels.

b) Stability performance is performed to evaluate and benchmark simulator dynamic stability (simulator drift) at 100% power.

c) Simulator instrument error is verified to be no greater than the Palo Verde Nuclear Generating Station Unit 1 plant while operating at 100% power.

3) Normal Operations Tests

a) This testing validates the capability to perform the normal plant evolutions of ANSI/ANS-3.5-1985, Section 3.1.1 by performing the relevant General Operating Procedures, System Operating Procedures, and Surveillance Procedures (and/or procedure sections).

b) Regulatory Guide 1.149, Revision 1, April 1987 requires that the normal plant evolutions of ANSI/ANS-3.5-1985, Section 3.1.1 be performed periodically (at the rate of approximately 25% per year) to verify continuing integrity of the simulator models.

c) Additionally, Emergency Operating, Abnormal Operating, and Alarm Response Procedures (or portions of those procedures) are performed as necessary to validate the capability for operators to respond to the specific malfunction events required by ANSI/ANS-3.5-1985, Section 3.1.2 and the transient events listed in ANSI/ANS-3.5-1985, Appendix B, Section B1.2.

This testing was performed by performance of simulator validation of any significant changes to plant Emergency Operating, Abnormal Operating, and Alarm Response Procedures conducted by the Operations Standards and the Operations Training Departments. Any simulator performance discrepancies were resolved via the simulator discrepancy reporting process.

Attachment 1 lists the year that the specific Normal Operations Tests (NETs) and Operations Surveillance Tests (OSTs) were performed.

4) Transient and Malfunction Testing

a) Testing is performed to evaluate agreement between the simulator response and predicted responses for malfunction events. The testing validates dynamic and logical fidelity for the event without regard to operator action.

b) The set of malfunctions events validated consists of the events of ANSI/ANS-3.5-1985, Section 3.1.2 and Appendix B, Section B1.2. This is performed periodically to meet the periodic testing requirements of ANSI/ANS-3.5-1985 and Regulatory Guide 1.149, Revision 1, April 1987.

Attachment 1 lists the year that the specific Transient and Malfunction Tests were performed.

C. Description of performance tests, if different, to be conducted on the simulation facility during the subsequent four year period.

1) No new tests are currently scheduled to be performed. However, the operations surveillance tests' (OST001 through OST010) descriptions have been changed to reflect the current surveillance test title and procedure number.

2) No tests are currently scheduled to be deleted from those contained on the previous test schedule.

D. Schedule for the conduct of approximately 25 percent of the performance tests per year for the subsequent four years.

1) Attachment 2 provides the schedule for certification testing to be performed over the next four-year period. Approximately 25% of all malfunction tests are performed annually.

6.0 Additional Testing Performed

A. **Benchmark Testing.** All significant transients that occurred at PVNGS since the initial certification of the simulators at PVNGS were re-enacted on the simulator. Critical parameters and sequence of events were compared between the actual plant data and the simulator response. Significant deviations between simulator and plant response were documented by the generation of simulator discrepancy reports to correct the simulator response.

B. **IC Maintenance Testing.** Eighteen simulator initial conditions (ICs) are maintained. These include various times in core life, plant modes and power levels. These ICs are tested and certified by performance of the mode specific surveillance tests, operating logs and verifying that the plant conditions for that IC are in accordance with plant procedures. Additionally, all 100%, 50% and 30% power ICs have the appropriate steady state heat balance tests performed, as described in Attachment 1.

7.0 Planned Simulator Upgrades

A. Thermo-Hydraulic Model

The thermo-hydraulic model for the reactor coolant system (RCS) and the steam generators will be upgraded to RELAP ™ in late 1998. This will provide an increased ability to model Pressurizer transient response, Steam Generator response, and reduced inventory operations of the RCS to include mid loop conditions.

B. Core Model

Concurrently with the thermo-hydraulic model upgrade, the core model software will be upgraded to NESTLE ™. This upgrade will allow for enhanced modeling of core axial shape index and azimuthal tilt. Additionally, this upgrade will allow for rapid inclusion of new plant core designs into the simulator model.

Attachment 1: Simulator Certification Test Performance

This attachment provides information relative to certification performance testing of the Palo Verde Nuclear Generating Station Control Room Simulators. This testing has been performed in accordance with requirements as stated in ANS/ANSI 3.5-1985 and Regulatory Guide 1.149 Revision 1, April 1987.

Deviations from original certification test schedule:

Test NET001 (Cold Shutdown to Hot Standby) was originally scheduled to be performed in 1997. It was moved up one year to 1996. NET009 (Core performance testing) was moved from 1996 to 1997 to coincide with the new core model that was installed in 1997.

Actual Test Performance

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Performed
A.3.1	MTI002	Computer Real Time Test	Annually
B.2.1(01)	SST-001	Simulator Heat Balance at 100% Power	Annually
B.2.1(02)	SST-002	Simulator Heat Balance at 50% Power	Annually
B.2.1(03)	SST-003	Simulator Heat Balance at 30% Power	Annually
B.2.1.(04)	SST-004	Simulator Stability for 60 Minutes at 100% Power	Annually
B.2.2(01)	TTP-001	Manual Reactor Trip	Annually
B.2.2(02)	TTP-002	Simultaneous Trip of All Feedwater Pumps	Annually
B.2.2(03)	TTP-003	Simultaneous Closure of All Main Steam Isolation Valves	Annually
B.2.2(04)	TTP-004	Simultaneous Trip of All Reactor Coolant Pumps	Annually
B.2.2(05)	TTP-005	Trip of Any Single Reactor Coolant Pump	Annually
B.2.2(06)	TTP-006	Main Turbine Trip (Maximum Power Level Which Does Not Result in Intermediate Reactor Trip)	Annually
B.2.2(07)	TTP-007	Maximum Rate Power Ramp (100%, Down to Approximately 75%, and Back Up to 100%)	Annually
B.2.2(08)	TTP-008	Maximum Size Reactor Coolant System Rupture Combined With Loss of All Offsite Power	Annually
B.2.2(09)	TTP-009	Maximum Size Unisolable Main Steam Line Rupture	Annually
B.2.2(10)	TTP-010	Slow Primary System Depressurization to Saturated Conditions Using Pressurizer Relief or Safety Valve Stuck Open. (Inhibit Activation of High Pressure Emergency Core Cooling Systems)	Annually
3.1.2(01)(b)-2	CVM04	Loss of Coolant: Outside Primary Containment	1994
3.1.1(08)(a)	NET007	Plant Shutdown from Rated Power to Hot Standby	1994
3.1.1(08)(a)	NET007	Cooldown from Hot Standby to Cold Shutdown Conditions	1994
3.1.1(10)a	OST001	41ST-1AF01, AFN-P01 Operability	1994

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Performed
3.1.1(10)b	OST002	41ST-1AF02, AFA-P01 Operability	1994
3.1.1(10)c	OST003	41ST-1AF03, AFB-P01 Operability	1994
3.1.2(12)(c)	RDM02	Control Rod Failure Including: c. Drifting Rods	1994
3.1.2(12)(d)	RDM02	Control Rod Failure Including: d. Rod Drops	1994
3.1.2(12)(e)	RDM02	Control Rod Failure Including: e. Misaligned Rods	1994
3.1.2(12)(a)	RDM03	Control Rod Failure Including: a. Stuck Rods	1994
3.1.2(12)(e)	RDM03	Control Rod Failure Including: e. Misaligned Rods	1994
3.1.2(12)(b)	RDM06	Control Rod Failure Including: b. Uncoupled Rods	1994
3.1.2(14)	THM04	Fuel Cladding Failure Resulting in High Activity Reactor Coolant or Off Gas and the Associated High Radiation Alarms	1994
3.1.2(01)(a)	THM06	Loss of Coolant: Significant PWR Steam Generator Leaks	1994
3.1.2(01)(b)-1	THM08	Loss of Coolant: Inside Primary Containment	1994
3.1.2(22)	ANOR01	Process Instrumentation, Alarms, and Control System Failures (Loss of Annunciators)	1995
3.1.2(18)(b)	CVM06	Failure of Reactor Coolant Volume Control Systems (PWR)	1995
3.1.2(22)	CVM14	Process Instrumentation, Alarms, and Control System Failures (PZR Level Transmitter Failure)	1995
3.1.2(20)(d)	FWM05	Main Feed Line Break Outside Containment	1995
3.1.2(20)(c)	FWM12	Main Feed Line Break Inside Containment	1995
3.1.2(09)	FWOR30	Loss of Normal Feedwater or Normal Feedwater System Failure	1995
3.1.2(20)(a)	MSM01	Main Steam Line Break Inside Containment	1995
3.1.2(22)	MSM02	Process Instrumentation, Alarms, and Control System Failures (FWCS Stem Flow Transmitter Failure)	1995
3.1.2(20)(b)	MSM07	Main Steam Line Break Outside Containment	1995
3.1.1(02)	NET002	Nuclear Startup - Hot Standby to Rated Power	1995
3.1.1(03)	NET003	Turbine Startup and Generator Synchronization	1995
3.1.1(04)	NET004	Reactor Tripped Followed by Recovery to Rated Power	1995
3.1.1(05)	NET005	Operations at Hot Standby. (Trip and Restart of a Reactor Coolant Pump)	1995
3.1.1(06)	NET006	Load Changes	1995
3.1.1(10)d	OST004	41ST-1DG01, DG "A" Operability Test	1995
3.1.1(10)e	OST005	41ST-1DG02, DG "B" Operability Test	1995
3.1.1(10)f	OST006	41ST-1RC02, RCS Water Inventory Balance	1995

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Performed
3.1.2(18)(a)	RCM05	Failure of Reactor Coolant Pressure Control Systems (PWR)	1995
3.1.2(19)	RPM05	Reactor Trip (RTSG Trip)	1995
3.1.2(11)	RXM02	Loss of Protective System Channel	1995
3.1.2(11)	RXM03	Loss of Protective System Channel	1995
3.1.2(11)	RXM04	Loss of Protective System Channel	1995
3.1.2(11)	RXM05	Loss of Protective System Channel	1995
3.1.2(22)	RXOR05	Process Instrumentation, Alarms, and Control System Failures (RRS Tc Input Failure)	1995
3.1.2(06)	TPM03	Loss of Service Water or Cooling to Individual Components	1995
3.1.2(09)	TTP013	Loss of Normal Feedwater or Normal Feedwater System Failure	1995
3.1.2(03)(a)-1	EDM02	Loss or Degraded Electrical Power to the Station: a. Loss of Offsite Power	1996
3.1.2(03)(d)-2	EDM02	Loss or Degraded Electrical Power to the Station: d. Loss of Power to the Plant's Electrical Distribution Buses	1996
3.1.2(03)(a)-2	EDM07	Loss or Degraded Electrical Power to the Station: a. Loss of Offsite Power	1996
3.1.2(03)(d)-1	EDM07	Loss or Degraded Electrical Power to the Station: d. Loss of Power to the Plant's Electrical Distribution Buses	1996
3.1.2(03)(b)	EDM11	Loss or Degraded Electrical Power to the Station: b. Loss of Emergency Power	1996
3.1.2(03)(d)-3	EDM11	Loss or Degraded Electrical Power to the Station: d. Loss of Power to the Plant's Electrical Distribution Buses	1996
3.1.2(03)(e)-1	EDM12	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	1996
3.1.2(03)(e)-1	EDM13	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	1996
3.1.2(03)(e)-2	EDM14	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	1996
3.1.2(03)(e)-2	EDM16	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	1996
3.1.2(03)(c)	EGM05	Loss or Degraded Electrical Power to the Station: c. Loss of Emergency Generators	1996
3.1.2(03)(c)	EGM06	Loss or Degraded Electrical Power to the Station: c. Loss of Emergency Generators	1996

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Performed
3.1.2(10)	FWM17	Loss of All Feedwater (Normal and Emergency Cooling)	1996
3.1.2(10)	FWM21	Loss of All Feedwater (Normal and Emergency Cooling)	1996
3.1.1(01)	NET001	Plant Startup - Cold Shutdown to Hot Standby	1996
3.1.1(09)(b)	NET009	Core Performance Testing Such As: Determination of Shutdown Margin	1996
3.1.1(09)(c)	NET009	Core Performance Testing Such As: Measurement of Reactivity Coefficients	1996
3.1.1(09)(d)	NET009	Core Performance Testing Such As: Measurement of Control Rod Worth	1996
3.1.1(10)g	OST007	41ST-1SG01, MSIV's Surveillance	1996
3.1.1(10)h	OST008	41ST-1SI03, Containment Spray Pump Operability Test	1996
3.1.2(15)	TCM13	Turbine Trip	1996
3.1.2(01)(c)-1	THM01	Loss of Coolant: Large Coolant Breaks Including Demonstration of Saturation Condition	1996
3.1.2(01)(c)-2	THM07	Loss of Coolant: Small Reactor Coolant Breaks Including Demonstration of Saturation Condition	1996
3.1.2(10)	TTP-011	Loss of All Feedwater (Normal and Emergency Cooling)	1996
3.1.2(08)	CCM03	Loss of Component Cooling System or Cooling to Individual Components	1997
3.1.2(23)(a)	CCM06	Passive Malfunctions in Systems, such as: a. Engineered Safety Features	1997
3.1.2(16)	EGM02	Generator Trips	1997
3.1.2(23)(a)	EGM09	Passive Malfunctions in Systems, such as: a. Engineered Safety Features	1997
3.1.2(23)(b)	FWM23	Passive Malfunctions in Systems, such as: b. Emergency Feedwater Systems	1997
3.1.2(02)	IAM05	Loss of Instrument Air	1997
3.1.2(05)(a)	MCM01	Loss of Condenser Vacuum Including Loss of Condenser Level Control	1997
3.1.2(05)(b)	MCM02	Loss of Condenser Vacuum Including Loss of Condenser Level Control	1997
3.1.1(09)(a)	NET009	Core Performance Testing Such As: Plant Heat Balance	1997
3.1.2(21)	NIM02	Nuclear Instrumentation Failures	1997
3.1.2(21)	NIM04	Nuclear Instrumentation Failures	1997
3.1.1(10)i	OST009	41ST-1SI10, HPSI Pump Operability	1997
3.1.1(10)j	OST010	41ST-1SI14, Section XI LPSI Test	1997
3.1.2(04)	RCM01	Loss of Forced Core Coolant Flow Due to Single or Multiple Pump Failure	1997

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Performed
3.1.2(17)(a)	RDM10	Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal	1997
3.1.2(13)	RDM11	Inability to Drive Control Rods	1997
3.1.2(07)	RHM03	Loss of Shutdown Cooling	1997
3.1.2.(24)	RPM04	Failure of the Automatic Reactor Trip System	1997
3.1.2(17)(b)	RXOR08	Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal	1997
3.1.2(01)(d)-1	THOR1	Loss of Coolant: Failure of Safety Valves	1997
3.1.2(01)(d)-2	THOR1	Loss of Coolant: Failure of Relief Valves	1997

Attachment 2: Simulator Certification Four-Year Test Schedule

This attachment provides the Palo Verde Nuclear Generating Station Control Room Simulator Certification Test schedule through the year 2001. This test schedule has been modified from the initial certification submittal to reflect the current surveillance test title and procedure number of surveillance tests OST-001 through OST-010.

Testing performed meets criteria as specified in ANSI/ANS - 3.5 - 1985 and Regulatory Guide 1.149 Revision 1. All malfunctions are scheduled to be completed on a four-year rotation with approximately 25% performed annually.

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Scheduled
A.3.1	MTI002	Computer Real Time Test	Annually
B.2.1(01)	SST-001	Simulator Heat Balance at 100% Power	Annually
B.2.1(02)	SST-002	Simulator Heat Balance at 50% Power	Annually
B.2.1(03)	SST-003	Simulator Heat Balance at 30% Power	Annually
B.2.1.(04)	SST-004	Simulator Stability for 60 Minutes at 100% Power	Annually
B.2.2(01)	TTP-001	Manual Reactor Trip	Annually
B.2.2(02)	TTP-002	Simultaneous Trip of All Feedwater Pumps	Annually
B.2.2(03)	TTP-003	Simultaneous Closure of All Main Steam Isolation Valves	Annually
B.2.2(04)	TTP-004	Simultaneous Trip of All Reactor Coolant Pumps	Annually
B.2.2(05)	TTP-005	Trip of Any Single Reactor Coolant Pump	Annually
B.2.2(06)	TTP-006	Main Turbine Trip (Maximum Power Level Which Does Not Result in Intermediate Reactor Trip)	Annually
B.2.2(07)	TTP-007	Maximum Rate Power Ramp (100%, Down to Approximately 75%, and Back Up to 100%)	Annually
B.2.2(08)	TTP-008	Maximum Size Reactor Coolant System Rupture Combined With Loss of All Offsite Power	Annually
B.2.2(09)	TTP-009	Maximum Size Unisolable Main Steam Line Rupture	Annually
B.2.2(10)	TTP-010	Slow Primary System Depressurization to Saturated Conditions Using Pressurizer Relief or Safety Valve Stuck Open. (Inhibit Activation of High Pressure Emergency Core Cooling Systems)	Annually
3.1.2(01)(b)-2	CVM04	Loss of Coolant: Outside Primary Containment	1998
3.1.1(08)(a)	NET007	Plant Shutdown from Rated Power to Hot Standby.	1998
3.1.1(08)(a)	NET007	Cooldown from Hot Standby to Cold Shutdown Conditions.	1998
3.1.1(10)a	OST001	73ST-9AF01, AFN-P01 In-Service Test	1998
3.1.1(10)b	OST002	73ST-9AF02, AFA-P01 In-Service Test	1998
3.1.1(10)c	OST003	73ST-9AF03, AFB-P01 In-Service Test	1998
3.1.2(12)(c)	RDM02	Control Rod Failure Including: c. Drifting Rods	1998

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Scheduled
3.1.2(12)(d)	RDM02	Control Rod Failure Including: d. Rod Drops	1998
3.1.2(12)(e)	RDM02	Control Rod Failure Including: e. Misaligned Rods	1998
3.1.2(12)(a)	RDM03	Control Rod Failure Including: a. Stuck Rods	1998
3.1.2(12)(e)	RDM03	Control Rod Failure Including: e. Misaligned Rods	1998
3.1.2(12)(b)	RDM06	Control Rod Failure Including: b. Uncoupled Rods	1998
3.1.2(14)	THM04	Fuel Cladding Failure Resulting in High Activity Reactor Coolant or Off Gas and the Associated High Radiation Alarms	1998
3.1.2(01)(a)	THM06	Loss of Coolant: Significant PWR Steam Generator Leaks	1998
3.1.2(01)(b)-1	THM08	Loss of Coolant: Inside Primary Containment	1998
3.1.2(22)	ANOR01	Process Instrumentation, Alarms, and Control System Failures (Loss of Annunciators)	1999
3.1.2(18)(b)	CVM06	Failure of Reactor Coolant Volume Control Systems (PWR)	1999
3.1.2(22)	CVM14	Process Instrumentation, Alarms, and Control System Failures (PZR Level Transmitter Failure)	1999
3.1.2(20)(d)	FWM05	Main Feed Line Break Outside Containment	1999
3.1.2(20)(c)	FWM12	Main Feed Line Break Inside Containment	1999
3.1.2(09)	FWOR30	Loss of Normal Feedwater or Normal Feedwater System Failure	1999
3.1.2(20)(a)	MSM01	Main Steam Line Break Inside Containment	1999
3.1.2(22)	MSM02	Process Instrumentation, Alarms, and Control System Failures (FWCS Stem Flow Transmitter Failure)	1999
3.1.2(20)(b)	MSM07	Main Steam Line Break Outside Containment	1999
3.1.1(02)	NET002	Nuclear Startup - Hot Standby to Rated Power	1999
3.1.1(03)	NET003	Turbine Startup and Generator Synchronization	1999
3.1.1(04)	NET004	Reactor Tripped Followed by Recovery to Rated Power	1999
3.1.1(05)	NET005	Operations at Hot Standby. (Trip and Restart of a Reactor Coolant Pump)	1999
3.1.1(06)	NET006	Load Changes	1999
3.1.1(10)d	OST004	40ST-9DG01, Diesel Generator A Test	1999
3.1.1(10)e	OST005	40ST-9DG02, Diesel Generator B Test	1999
3.1.1(10)f	OST006	40ST-9RC02, Computer Calculation of RCS Water Inventory	1999
3.1.2(18)(a)	RCM05	Failure of Reactor Coolant Pressure Control Systems (PWR)	1999
3.1.2(19)	RPM05	Reactor Trip (RTSG Trip)	1999

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Scheduled
3.1.2(11)	RXM02	Loss of Protective System Channel	1999
3.1.2(11)	RXM03	Loss of Protective System Channel	1999
3.1.2(11)	RXM04	Loss of Protective System Channel	1999
3.1.2(11)	RXM05	Loss of Protective System Channel	1999
3.1.2(22)	RXOR05	Process Instrumentation, Alarms, and Control System Failures (RRS Tc Input Failure)	1999
3.1.2(06)	TPM03	Loss of Service Water or Cooling to Individual Components	1999
3.1.2(09)	TTP013	Loss of Normal Feedwater or Normal Feedwater System Failure	1999
3.1.2(03)(a)-1	EDM02	Loss or Degraded Electrical Power to the Station: a. Loss of Offsite Power	2000
3.1.2(03)(d)-2	EDM02	Loss or Degraded Electrical Power to the Station: d. Loss of Power to the Plant's Electrical Distribution Buses	2000
3.1.2(03)(a)-2	EDM07	Loss or Degraded Electrical Power to the Station: a. Loss of Offsite Power	2000
3.1.2(03)(d)-1	EDM07	Loss or Degraded Electrical Power to the Station: d. Loss of Power to the Plant's Electrical Distribution Buses	2000
3.1.2(03)(b)	EDM11	Loss or Degraded Electrical Power to the Station: b. Loss of Emergency Power	2000
3.1.2(03)(d)-3	EDM11	Loss or Degraded Electrical Power to the Station: d. Loss of Power to the Plant's Electrical Distribution Buses	2000
3.1.2(03)(e)-1	EDM12	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	2000
3.1.2(03)(e)-1	EDM13	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	2000
3.1.2(03)(e)-2	EDM14	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	2000
3.1.2(03)(e)-2	EDM16	Loss or Degraded Electrical Power to the Station: e. Loss of Power to the Individual Instrument Buses (AC as well as DC)	2000
3.1.2(03)(c)	EGM05	Loss or Degraded Electrical Power to the Station: c. Loss of Emergency Generators	2000
3.1.2(03)(c)	EGM06	Loss or Degraded Electrical Power to the Station: c. Loss of Emergency Generators	2000
3.1.2(10)	FWM17	Loss of All Feedwater (Normal and Emergency Cooling)	2000
3.1.2(10)	FWM21	Loss of All Feedwater (Normal and Emergency Cooling)	2000

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Scheduled
3.1.1(09)(a)	NET009	Core Performance Testing Such As: Plant Heat Balance	2000
3.1.1(09)(b)	NET009	Core Performance Testing Such As: Determination of Shutdown Margin	2000
3.1.1(09)(c)	NET009	Core Performance Testing Such As: Measurement of Reactivity Coefficients	2000
3.1.1(09)(d)	NET009	Core Performance Testing Such As: Measurement of Control Rod Worth	2000
3.1.1(10)g	OST007	73ST-9SG01, MSIV In-Service Test	2000
3.1.1(10)h	OST008	73ST-9SI06, Containment Spray Pumps In-Service Test	2000
3.1.2(15)	TCM13	Turbine Trip	2000
3.1.2(01)(c)-1	THM01	Loss of Coolant: Large Coolant Breaks Including Demonstration of Saturation Condition	2000
3.1.2(01)(c)-2	THM07	Loss of Coolant: Small Reactor Coolant Breaks Including Demonstration of Saturation Condition	2000
3.1.2(10)	TTP-011	Loss of All Feedwater (Normal and Emergency Cooling)	2000
3.1.2(08)	CCM03	Loss of Component Cooling System or Cooling to Individual Components	2001
3.1.2(23)(a)	CCM06	Passive Malfunctions in Systems, such as: a. Engineered Safety Features	2001
3.1.2(16)	EGM02	Generator Trips	2001
3.1.2(23)(a)	EGM09	Passive Malfunctions in Systems, such as: a. Engineered Safety Features	2001
3.1.2(23)(b)	FWM23	Passive Malfunctions in Systems, such as: b. Emergency Feedwater Systems	2001
3.1.2(02)	IAM05	Loss of Instrument Air	2001
3.1.2(05)(a)	MCM01	Loss of Condenser Vacuum Including Loss of Condenser Level Control	2001
3.1.2(05)(b)	MCM02	Loss of Condenser Vacuum Including Loss of Condenser Level Control	2001
3.1.1(01)	NET001	Plant Startup - Cold Shutdown to Hot Standby	2001
3.1.2(21)	NIM02	Nuclear Instrumentation Failures	2001
3.1.2(21)	NIM04	Nuclear Instrumentation Failures	2001
3.1.1(10)i	OST009	73ST-9SI10, HPSI Pumps Miniflow In-Service Test	2001
3.1.1(10)j	OST010	73ST-9SI11, Low Pressure Safety Injection Pumps Miniflow In-Service Test	2001
3.1.2(04)	RCM01	Loss of Forced Core Coolant Flow Due to Single or Multiple Pump Failure	2001
3.1.2(17)(a)	RDM10	Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal	2001

ANSI 3.5 Paragraph Number	PVNGS Test Number	Test Description	Year Scheduled
3.1.2(13)	RDM11	Inability to Drive Control Rods	2001
3.1.2(07)	RHM03	Loss of Shutdown Cooling	2001
3.1.2.(24)	RPM04	Failure of the Automatic Reactor Trip System	2001
3.1.2(17)(b)	RXOR08	Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal	2001
3.1.2(01)(d)-1	THOR1	Loss of Coolant: Failure of Safety Valves	2001
3.1.2(01)(d)-2	THOR1	Loss of Coolant: Failure of Relief Valves	2001