Robert G. Byram
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SEP 25 2001

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
NRC FORM 474 "SIMULATION FACILITY CERTIFICATION"
PLA-5371

Docket No. 50-387
and 50-388

In accordance with 10CFR55.45(b)(5)(ii) PPL Susquehanna, LLC submits NRC Form 474 "Simulation Facility Certification" for the Susquehanna Steam Electric Station Unit 1 and Unit 2. These certifications support changes in simulator performance tests conducted since the original certification (PLA-4025, dated September 28, 1993) and the previous certification (PLA-4671, dated September 17, 1997).

Please contact Mr. Walter W. Hunt, (570-542-3619), if you need additional information or clarification.

Sincerely,

R. G. Byram

Attachment

Copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. Schaaf, NRC Project Manager

A001

2001 QUADRENIAL SIMULATOR CERTIFICATION TEST REPORT

Submitted By:

, 9/14/01
Test Director

Reviewed By:

, 9/14/01
Supervisor-Nuclear Instruction Operations

Approved By:

, 9/17/01
Manager-Nuclear Training

Susquehanna Steam Electric Station, Unit One **Docket # 50 - 387**

Susquehanna Steam Electric Station, Unit Two **Docket #50 388**

PPL Susquehanna, LLC **September 14, 2001**

QUADRENNIAL CERTIFICATION REPORT
INDEX

Summary

Performance Tests Conducted 1998-2001

Annual performance tests for the years 1998 through 2001 were completed as scheduled in the initial Certification Report.

Performance Tests Discrepancies 1998 – 2001

Discrepancies documented during the conduct of the performance tests from 1998 through 2001 have been corrected with the exception of certification Test Discrepancies S5980241. Test Discrepancy S5980241 identified during the performance of Test 5110 (08/25/97) is scheduled to be completed by 12/31/97.

Performance Test Corrections 1998 – 2001

Performance test corrections are included in this report. As necessary, test corrections were made and the tests rerun to reflect model or procedural enhancements.

Performance Test Schedule 1998 – 2001

The performance test schedule for the next four years is the same as the original schedule in the Initial Certification Report and is reproduced in this report.

NRC FORM 474 **U.S. NUCLEAR REGULATORY COMMISSION**
(8-1998)

SIMULATION FACILITY CERTIFICATION

APPROVED BY OMB: NO. 3150-0138EXPIRES: 08/31/2001

Estimated burden per response to comply with this mandatory information collection request: 120 hours. This information is used to certify a simulation facility. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0138), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

INSTRUCTIONS: This form is to be filed for initial certification, recertification (if required), and for any change to a simulation facility performance testing plan made after initial submittal of such a plan. Provide the following information and check the appropriate box to indicate reason for submittal.

Susquehanna Steam Electric Station – Unit 1	DOCKET NUMBER 50-387
LICENSEE PPL Susquehanna, LLC	DATE 09/14/01

This is to certify that:

1. The above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR 55.45.
2. Documentation is available for NRC review in accordance with 10 CFR 55.45(b).
3. This simulation facility meets the guidance contained in ANSI/ANS 3.5-1985 or ANSI/ANS 3.5-1993, as endorsed by NRC Regulatory Guide 1.149.

If there are any **EXCEPTIONS** to the certification of this item, **CHECK HERE** [X] and describe fully on additional pages as necessary.

NAME *(or other identification)* AND LOCATION OF SIMULATION FACILITY.

SSES Simulator
Route 11, Approximately Five Miles North of Berwick, PA

X SIMULATION FACILITY PERFORMANCE TEST ABSTRACTS ATTACHED. *(For performance tests conducted in the period ending with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING COMPLETED. *(Attach additional pages as necessary and identify the item description being continued.)*

See Page 6 for Performance Test Corrections made to performance Tests Conducted Since Original Certification.

X SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. *(For the conduct of approximately 25 percent of performance tests per year for the four-year period commencing with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING TO BE CONDUCTED. *(Attach additional pages as necessary and identify the item description being continued.)*

X PERFORMANCE TESTING PLAN CHANGE. *(For any modification to a performance testing plan submitted on a previous certification.)*

DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE *(Attach additional pages as necessary and identify the item description being continued.)*

[ ] RECERTIFICATION *(Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR 55.45(b)(5)(v). (Attach additional pages as necessary and identify the item description being continued.)*

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE AUTHORIZED REPRESENTATIVE	TITLE	DATE
[signature]	Senior Vice President and Chief Nuclear Officer	09/14/01

In accordance with 10 CFR 55.5, Communications, this form shall be submitted to the NRC as follows:
BY MAIL ADDRESSED TO: **DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION U.S. NUCLEAR REGULATORY COMMISSION WASHINGTON, DC 20555-0001**

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: **ONE WHITE FLINT NORTH 11555 ROCKVILLE PIKE ROCKVILLE, MD**

Susquehanna Steam Electric Station, Unit One

Docket # 50 - 387

PPL Susquehanna, LLC

September 14, 2001

EXCEPTIONS

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT ONE

PPL SUSQUEHANNA, LLC

UNIT ONE CERTIFICATION EXCEPTIONS ARE AS LISTED IN THE ORIGINAL CERTIFICATION DOCUMENT DATED SEPTEMBER 27, 1993.

SUSQUEHANNA STEAM ELECTRIC STATION
PPL SUSQUEHANNA, LLC
SEPTEMBER 14, 2001

PERFORMANCE TEST CORRECTIONS

- Test 6501 was revised to correct a math error and to use ten minute averages for Feedwater temperature because these points utilize more significant digits and are more representative of steady state performance. The compression limits errors are also substantially eliminated, also the deadband of 3% for Feedwater was minimized to stay within our 2% criteria per ANSI Standard.

- Two tests 5202, and 5203 were revised to incorporate changes to the EOP's since the test was last conducted.

- Two tests 5207 and 5218 were revised due to changes in the instrument air system alignment. Also the malfunction for 5207 test was corrected to use proper break dynamics for a double ended guillotine break.

- One test 5215 was corrected to show proper alarm to check on a Bus Lockout.

- Test 5232 was corrected to use current EOP level of -110" and HPCI initiation set point of -38".

- Test 5503 was revised to Removed reference to DCS formats for data. Will use only computer point designations since Plant computer was upgraded to PICSY all screen format names are changed but point designations are the same.

- Test 5106 was changed to use GO-100-005 Plant Shutdown to Cold Shutdown from GO-100-011 Plant Shutdown Following a Scram.

- Test 5108 was changed to reflect use of Data from current core in simulator model. Startup and Operations Report for Unit 1 Cycle 10 core as compiled by Nuclear Fuels (PL-NF-96-006).

- Test 5109, Step 5 was changed to Read Ensure the current BOL Startup sequence is selected. Added reference to use the Nuclear Fuels Startup and Operations report for the currently installed core model.

- Test 5103 was revised to Change the Startup Sequence to A2 and reference the Remote Function JS156004 for this sequence. This change was due to the Unit 1 Cycle 10 Core Load and the New RWM PICSY program.

- Test 5214 was changed to remove reference to 1C644 Panel Loss of Reactor Vessel Pressure indication PI-E32-1R660. This was removed under a Modification to remove MSIV Leakage Control from the Plant.

PERFORMANCE TEST CORRECTIONS CONTINUED

- Test 5253 was changed to correct a problem with the 4 kV breaker tripping on undervoltage; that was correct with a CSPR.
- Test 5308 reflects a modification on the 0C659 panel that moved the alarms for the MOAB's.
- Test 6503 was revised to reincorporate Exception 006 for Reactor Recirc Loop's "A" and "B" Drive Flow.
- Test 5110 was revised to remove procedures for systems that are now removed in the plant and procedure names or numbers that were changed.
- Test 5228 was revised to incorporate a change from OP-173-001 to OP-173-003.
- Test 5243 was revised to incorporate data from PLI-0086781 for level changes and incorporate Control of RCIC per procedure.

Listing of Certification Test Deficiencies

Simulator deficiencies noted during the performance of ANS-3.5 testing that have not been corrected at the time of submittal are listed below. Correction Action for each deficiency is to modify the simulator as necessary to meet the ANSI/ANS 3.5-1985 acceptance criteria Projected dates of completion for each of the deficiencies are listed below as well as the number of the deficiency.

The overall dynamic results of the testing is satisfactory; however, until the deficiencies have been corrected, the deficiencies that could have an undesired impact on the student can be overridden by the instructor to prevent negative training.

Number	Anticipated Completion	Deficiency
S5010147	03/11/02	ESW Quarterly Flow surveillances SO-054-A02 and SO-054-B02

Susquehanna Steam Electric Station, Unit One

PPL Susquehanna, LLC

September 14, 2001

PERFORMANCE TESTS COMPLETED

Simulator Tests Conducted Annually:

5610	Appendix A3.1 Simulator Computer Real Time Test
5501	Appendix B1.1 Steady-State Performance – 30 Percent Power
5502	Appendix B1.1 Steady-State Performance – 75 Percent Power
5503	Appendix B1.1 Steady-State Performance – 100 Percent Power and One Hour Stability
6501	Appendix B1.1 Unit Two Steady-State Performance – 30 Percent Power
6502	Appendix B1.1 Unit two Steady-State Performance – 70 Percent Power
6503	Appendix B1.1 Unit Two Steady-State Performance – 100 Percent Power
5301	Appendix B1.2(1) Transient Performance-Manual Scram
5302	Appendix B1.2(2) Transient Performance – Simultaneous Trip of All Feedwater Pumps
5303	Appendix B.12(3) Transient Performance – Simultaneous Closer of All Main Steam Isolation Valves
5304	Appendix B1.2(4) Transient Performance – Simultaneous Trip of All Recirculation Pumps
5305	Appendix B1.2(5) Transient Performance – Single Recirculation Pump Trip
5306	Appendix B1.2(6) Transient Performance – Main Turbine Trip (Maximum Power Level Which Does Not Result in an Immediate Reactor Scram)
5307	Appendix B1.2(7) Transient Performance – Maximum Power Ramp (Master Recirc Controller in Manual) Down to Approximately 75 Percent and Back Up to 100 Percent
5308	Appendix B1.2(8) Transient Performance – Maximum Size Reactor Coolant System Rupture Combined With a Loss of Offsite Power
5309	Appendix B1.2(9) Transient Performance – Maximum Size Unisolable Main Steam Line Rupture
5310	Appendix B1.2(10) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves Combined With Single Stuck Open Safety/Relief Valve (Inhibit Activation of High Pressure ECCS)

Additional Simulator Tests Conducted the First Year:

5601	Section 4.3 Simulator Operating Limits (Drywell Floor dp Down and High Reactor Pressure)
5101	3.1.1(1.5) Plant Startup – Cold to Hot Standby (Including Operations at Hot Standby Startup)
5102	3.1.1(2,3) Nuclear Startup from Hot Standby to Rated Power (including Turbine Startup and Generator Synchronization)
5202	3.12(1) Small Reactor Coolant Breaks Outside Primary Containment ("D" Main Steam Line Leak Outside Containment TB Pipe Tunnel at Two Percent Severity)
5203	3.12(1) Small Reactor Coolant Breaks Inside Primary Containment (Reactor Vessel Bottom Head Drain Line Leakage)
5205	3.12(2) Loss of Instrument Air (Instrument Air Leakage – Turbine Building Elevation 646')
5206	3.12(2) Loss of Instrument Air (Instrument Air Common Header Rupture)
5207	3.12(2) Loss of Instrument Air (Instrument Air Leakage – Turbine Building Elevation 699')
5215	3.12(3) Loss or Degraded Electrical Power to the Station (Startup Bus 10 Lockout)
5216	3.12(3) Loss or Degraded Electrical Power to the Station (Startup Buses 10 and 20 Lockout and Generator Synch Output Breaker 1R101 Spurious Trip)
5217	3.12(3) Loss or Degraded Electrical Power to the Station (Auxiliary Bus 11A – Fails to Fast Transfer)
5218	3.12(3) Loss or Degraded Electrical Power to the Station (4.16 kV ESS Bus 18 Differential Current Lockout Trip and Diesel Generator "B: Fails to Start)
5219	3.12(3) Loss or Degraded Electrical Power to the Station (13.8 kV Bus 11B Lockout)
5224	3.12(7) Loss of Shutdown Cooling (RHRSW Loop 1A Leak/Rupture)
5229	3.12(9) Normal Feedwater System Failure (Reactor Feed Pump C Trip)
5230	3.12(9) Normal Feedwater System Failure (Loss of Extraction Steam to Feedwater Heater 4B)

5231	3.12(10) Loss of All Feedwater (Normal and Emergency) (RCIC Turbine Trip, HPCI Auto Start Failure, Reactor Feed Pumps A, B, and C Trip
5232	3.12(11) Loss of Protective System Channel (Failure to Scram – Low Power Natural Circulation Mists Closed ATWS)
5233	3.12(11) Loss of Protective System Channel (Failure to Scram – High Power Mists Open ATWS)
5239	3.12(15) Turbine Trip (Main Turbine Trip)
5240	3.12(16) Generator Trip (Negative Phase Sequence Trip – Generator Load Reject)
5245	3.12(20) Main Feed Line Break Outside Containment (Feedwater Discharge Line "A" Leak Outside Containment)
5259	3.12(25) Reactor Pressure Control System Failure Including Turbine Bypass Failure (Loss of Turbine Primary and Backup Speed Signals)
5260	3.12(25) Reactor Pressure Control System Failure Including Turbine Bypass Failure (Loss of 24 VDC Logic Bus Power)
5261	3.12(25) Reactor Pressure Control System Failure (Turbine Pressure Regulator Oscillation)

Simulator Tests Conducted Annually:

5610	Appendix A3.1 Simulator Computer Real Time Test
5501	Appendix B1.1 Steady-State Performance – 30 Percent Power
5502	Appendix B1.1 Steady-State Performance – 75 Percent Power
5503	Appendix B1.1 Steady-State Performance – 100 Percent Power and One Hour Stability
6501	Appendix B1.1 Unit Two Steady-State Performance – 30 Percent Power
6502	Appendix B1.1 Unit Two Steady-State Performance – 70 Percent Power
6503	Appendix B1.1 Unit Two Steady-State Performance – 100 Percent Power
5301	Appendix B1.2 Transient Performance – Manual Scram
5302	Appendix B1.2(2) Transient Performance – Simultaneous Trip of All Feedwater Pumps
5303	Appendix B1.2(3) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves
5304	Appendix B1.2(4) Transient Performance – Simultaneous Trip of All Recirculation Pumps
5304	Appendix B1.2(5) Transient Performance – Single Recirculation Pump Trip
5306	Appendix B1.2(6) Transient Performance – Main Turbine Trip (Maximum Power Level Which Does Not Result in and Immediate Reactor Scram)
5307	Appendix B1.2(7) Transient Performance – Maximum Power Ramp (Master Recirc Controller in Manual Down to Approximately 75 Percent and Back Up to 100 Percent
5308	Appendix B1.2(8) Transient Performance – Maximum Size Reactor Coolant System Rupture Combined With a Loss of Offsite Power
5309	Appendix B1.2(9) Transient Performance – Maximum Size Unisolable Main Steam Line Rupture
5310	Appendix B1.2(10) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves Combined With Single Stuck Open Safety/Relief Valve (Inhibit Activation of High Pressure ECCS)

Additional Simulator Tests Conducted the Second Year:

5602	Section 4.3 Simulator Operating Limits (High Containment Pressure and Temperature, External Pressure, and Differential Pressure (UP))
5106	3.1.1(8, 6) Plant Shutdown from Rated Power to Hot Standby (including Load Changes) Followed by a Plant Cooldown to Cold Shutdown Conditions
5108	3.1.1(9) Shutdown Margin Determination
5109	3.1.1(9) In-Sequence Critical and Shutdown Margin Demonstration
5201	3.1.2(1) Large Reactor Control Breaks Outside Primary Containment ("D" Main Steam Line Leak Outside Containment TB Pipe Tunnel at 100 Percent Severity)
5204	3.1.2(1) Failure of Safety Relief Valves (Safety Relief Valves (Safety Relief Valve "M" Fails Open))
5208	3.1.2(3) Loss or Degraded Electrical Power to the Station (Sustained Station Blackout With Generator Output Breaker Trip)
5220	3.1.2(5) Loss of Condenser Vacuum Including Loss of Condenser Level Control (Main Condenser Tube Leak at One Percent Severity)
5221	3.1.2(5) Loss of Condenser Vacuum Including Loss of Condenser Level Control (Main Condenser Water Level Control Failure – High)
5222	3.1.2(5) Loss of Condenser Vacuum Including Loss of Condenser Level Control (Circ Water Pumps A, B, C and D Trip)
5223	3.1.2(6) Loss of Service Water (Service Water Pumps A, B. and C Trip)
5234	3.1.2(12) Control Rod Failure Including Drifting Rods and Misaligned Rods (Control Rod 22-27 Drift In)
5235	3.1.2(12) Control Rod Failure Including Drifting Rods and Misaligned Rods (Control Rod 26-35 Drift Out)
5236	3.1.2(12) Control Rod Failure Including Stuck Rods, Uncoupled Rods, and Dropped Rod (Control Rod 26-35 Stuck and Uncoupled)
5237	3.1.2(12) Inability to Drive Control Rods (RMCS Sequence Timer Malfunction)
5238	3.12(14) Fuel Cladding Failure Resulting in High Activity in Reactor Coolant (Fuel Cladding Failure)

5241	3.1.2(17) Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal (Actuation of the Recirc "B: 30 Percent Limiter)
5242	3.1.2(17) Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal (Inadvertent (Spurious) HPCI Initiation)

Simulator Tests Conducted Annually:

5610	Appendix A3.1 Simulator Computer Real Time Test
5501	Appendix B1.1 Steady-State Performance – 30 Percent Power
5502	1 Steady-State Performance – 75 Percent Power
5503	Appendix B1.1 Steady-State Performance – 100 Percent Power and One Hour Stability
6501	Appendix B1.1 Unit Two Steady-State Performance – 30 Percent Power
6502	Appendix B1.1 Unit Two Steady-State Performance – 70 Percent Power
6503	Appendix B1.1 Two Steady-State Performance – 100 Percent Power
5301	Appendix B1.2(1) Transient Performance – Manual Scram
5302	Appendix B1.2(2) Transient Performance – Simultaneous Trip of All Feedwater Pumps
5303	Appendix B1.2(3) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves
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5309	Appendix B1.2(9) Transient Performance – Maximum Size Unisolable Main Steam Line Rupture
5310	Appendix B1.2(10) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves Combined With Single Stuck Open Safety/Relief Valve (Inhibit Activation of High Pressure ECCS)

Additional Simulator Tests Conducted the Third Year:

5603	Section 4.3 Simulator Operating Limits (Fuel Clad Temperature)
5103	3.1.1(4) Reactor Trip Followed by Recovery to Rated Power
5107	3.1.1(9) Plant Heat Balance
5209	3.1.2(3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D612)
5210	3.1.2(3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D622)
5211	3.1.2(3) Loss of Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D632)
5212	3.1.2(3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D642)
5213	3.1.2(3) Loss or Degraded Electrical Power to the Station (Loss of AC Instrument Bus 1Y218/1Y219)
5214	3.1.2(3) Loss or Degraded Electrical Power to the Station Loss of AC Instrument Bus 1Y226)
5250	3.1.2(22a) Process Instrumentation System Failure (Recirc Flow Unit A Failure)
5251	3.1.2(22b) Process Alarm System Failure (Loss of Annunciator 1C651 - AR101)
5252	3.1.2(22b) Process Alarm System Failure (Loss of Annunciator 1C651 - AR110)
5253	3.1.2(22b) Process Alarm System Failure (Loss of Annunciator) C653 - AR015
5254	3.1.2(22c) Process Control System Failure (Recirc Pump A M/A Station Auto Output Failure – High)
5255	3.1.2(22c) Process Control System Failure (Feedwater Master Level Controller Failure – High)
5256	3.1.2(23) Passive Malfunction in Emergency Feedwater (Loss of Core Spray Division 2 Initiation Logic Power and Recir Loop B Discharge Line Break)
5257	3.1.2(23) Passive Malfunction in Emergency Feedwater (RCIC and HPCI Automatic Start Failure When All Feedwater Pumps Trip)
5258	3.1.2(24) Failure of the Automatic Reactor Trip System (Failure to Scram on a Loss of All Feedwater Pumps)

Simulator Tests Conducted Annually:

5610	Appendix A3.1 Simulator Computer Real Time Test
5601	Appendix B1.1 Steady-State Performance – 30 Percent Power
5502	Appendix B1.1 Steady-State Performance – 75 Percent Power
5503	Appendix B1.1 Steady-State Performance – 100 Percent Power
6501	Appendix B1.1 Unit Two Steady-State Performance – 30 Percent Power
6502	Appendix B1.1 Unit Two Steady-State Performance – 70 Percent Power
6503	Appendix B1.1 Unit Two Steady-State Performance – 100 Percent Power
5301	Appendix B1.2(1) Transient Performance – Manual Scram
5302	Appendix B1.292) Transient Performance – Simultaneous Trip of All Feedwater Pumps
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5309	Appendix B1.2(9) Transient Performance – Maximum Size Unisolable Main Steam Line Rupture
5310	Appendix B1.2(10) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves Combined With Single Stuck Open Safety/Relief Valve (Inhibit Activation of High Pressure ECCS)

Additional Simulator Tests Conducted the Fourth Year:

5604	Section 4.3 Simulator Operating Limits (250 V, 125 V, and 24 VDC Buses)
5103	3.1.(7a, c) Startup and Power Operations With Less Than Full Reactor Coolant Flow
5104	3.1.(7b, c) Shutdown and Power Operations With Less Than Full Reactor Coolant
5110	3.1.(10) Operator Conducted Surveillance Testing on Safety-Related Equipment or Systems
5225	3.1.2(8) Loss of Component Cooling System or Cooling to Individual Components (CRD Pump A Bearing and Lube Oil Cooler Flow Blockage)
5226	3.1.2(8) Loss of Component Cooling System or Cooling to Individual Components (Reactor Recirculation "A" M/G Set Hydraulic Fluid Cooler Flow Blockage)
5227	3.1.2(8) Loss of Component Cooling System or Cooling to Individual Components (Trip of All Drywell Cooling Fans)
5228	3.1.2(8) Loss of Component Cooling System or Cooling to Individual Components (Trip of All Drywell Cooling Fans)
5243	3.1.2(19) Reactor Trip (Control Rods in Scram Group No. 1 Receive a Scram Signal 45 Control Rods Scrammed)
5244	3.1.2(20) Main Feed Line Break Inside Containment (Feedwater Discharge Line "B" Leak/Rupture Inside Containment)
5246	3.1.2(21) Nuclear Instrumentation Failure (SRM Channel A Failure)
5247	3.1.2(21) Nuclear Instrumentation Failure (IRM Channel B Failure)
5248	3.1.2(21) Nuclear Instrumentation Failure (APRM Channel C Failure)
5249	3.1.2(21) Nuclear Instrumentation Failure (Rod Block Monitor Channel A Failure)

PERFORMANCE TEST ABSTRACTS/ANSA CROSS –REFERENCE

The following list of performance tests provide a cross-reference between the title of the performance test, the test number assigned by PPL Susquehanna, LLC, and the specific ANSI/ANS 3.5 Section of the test addresses. The Test Abstracts included in this attachment provide a description of the specific tests performed to show compliance with ANSI/ANS 3.5 – 1985 and Regulatory Guide 1.149.

The simulator malfunctions tested for certification are listed on the Test Abstracts. Malfunctions listed as tested at a particular severity are variable severity malfunctions and specify the severity used for the test.

The baseline data used in the evaluation of the simulator response to the test include, but are not limited to reference plant GETARs data, calculated RETRAN data, Event Summary Reports, and Significant Operating Occurrence Reports. Additionally, applicable reference plant Normal, Off-Normal, Emergency, and Surveillance procedures were utilized as baseline data for evaluations.

Performance Test Abstracts
Normal Operating Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5101	3.1.1	(1.5)	Plant Startup – Cold to Hot Standby (Including Operations at Hot Standby)
5102	3.1.1	(2, 3)	Nuclear Startup from Hot Standby to Rated Power (Including Turbine Startup and Generator Synchronization)
5103	3.1.1	(4)	Reactor Trip Followed by Recovery to Rated Power
5104	3.1.1	(7a, c)	Startup and Power Operation With Less Full Reactor Coolant Flow
5105	3.1.1	(7b, c)	Shutdown and Power Operations With Less Than Full Reactor Coolant Flow
5106	3.1.1	(8, 6)	Plant Shutdown from Rated Power to hot standby (including load changes) followed by a plan cooldown to cold shutdown conditions
5107	3.1.1	(9a)	Plant Heat Balance
5108	3.1.1	(9b)	Shutdown Margin Determination
5109	3.1.1	(9b)	In-Sequence Critical and Shutdown Margin Demonstration
N/A	3.1.1	(9c)	Measurement of reactivity coefficients (Certification Exception Number: __003__)
N/A	3.1.1	(9d)	Measurement of control rod worth (Certification Exception Number: __003__)
5110	3.1.1	(10)	Operator Conducted Surveillance Testing on Safety-Related Equipment or Systems

Normal Operating Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
N/A	3.1.2	(1a)	Significant PWR steam generator leaks (Certification Exception Number: 001 .)
Note	3.1.2	(1b, c)	Large reactor coolant breaks inside primary containment (Note: Conducted under Transient Performance Test 5308, maximum size reactor coolant system rupture combined with a loss of offsite power)
5201	3.1.2	(1b, c)	Large reactor coolant breaks outside primary containment (D Main Steam Line Leak Outside Containment pipe tunnel at 100 percent severity)
5202	3.1.2	(1b, c)	Small reactor coolant breaks outside primary containment (D Main Steam Line Leak Outside Containment TB pipe tunnel at two percent severity)
5203	3.1.2	(1b, c)	Small reactor coolant breaks inside primary containment ((Reactor Vessel Bottom Head Drain Line Leakage)
5204	3.1.2	(1d)	Failure of safety relief valves (Safety Relief Valve "M" fails open)
5205	3.1.2	(2)	Loss of instrument air (Instrument Air Leakage – Turbine Building Elevation 646')
5206	3.1.2	(2)	Loss of instrument air (Instrument Air Common Header Rupture)
5207	3.1.2	(2)	Loss of instrument air (Instrument Air Leakage – Turbine Building Elevation 699')
5208	3.1.2	(3)	Loss or degraded electrical power to the Station (sustained Station Blackout with Generator output breaker trip)
5209	3.1.2	(3)	Loss or degraded electrical power to the Station (Loss of 125 VDC Bus 1D612)
5210	3.1.2	(3)	Loss or degraded electrical power to the Station (Loss of 125 VDC Bus 1D622)

Plant Malfunction Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5211	3.1.2	(3)	Loss or degraded electrical power to the Station (Loss of 125 VDC Bus 1D632)
5212	3.1.2	(3)	Loss or degraded electrical power to the Station (Loss of 125 VDC Bus 1D642)
5213	3.12	(3)	Loss or degraded electrical power to the Station (Loss of AC Instrument Bus 1Y218/1Y219
5214	3.1.2	(3)	Loss or degraded electrical power to the Station (Loss of AC Instrument Bus 1Y226)
5215	3.1.2	(3)	Loss or degraded electrical power to the Station (Startup Bus 10 Lockout)
5216	3.1.2	(3)	Loss or degraded electrical power to the Station (Startup Buses 10 and 20 Lockout and Generator Synch Output Breaker 1R101 Spurious Trip)
5217	3.1.2	(3)	Loss or degraded electrical power to the Station (Auxiliary Bus 11A Fails to Fast Transfer)
5218	3.1.2	(3)	Loss or degraded electrical power to the Station (4.16 kV ESS Bus 1B Differential Current Lockout Trip and Diesel Generator B Fails to Start)
5219	3.1.2	(3)	Loss or degraded electrical power to the Station (13.8 kV Bus 11B Lockout)
Note	3.1.2	(4)	Loss of forced core coolant flow due to single pump failure (Note: Conducted under Transient Performance Test 5305, Single recirculation pump trip)
5220	3.1.2	(5)	Loss of condenser vacuum including loss of condenser level control (Main Condenser Tube leak at one percent severity)
5221	3.1.2	(5)	Loss of condenser vacuum including loss of condenser level control (Main Condenser Water Level Control Failure)

Plant Malfunction Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5222	3.1.2	(5)	Loss of condenser vacuum (Circ Water Pumps A, B, C, and D trip
5223	3.1.2	(6)	Loss of service water (Service Water Pumps A, B, and C Trip)
5224	3.12	(7)	Loss of shutdown cooling (RHRSW Loop 1A Leak Rupture)
5225	3.1.2	(8)	Loss of component cooling system or cooling to individual components (CRD Pump A Bearing and Lo Cooler Flow Blockage)
5226	3.1.2	(8)	Loss of component cooling system or cooling to individual components (Reactor Recirculation 'A' M/G Set Hydraulic Fluid Cooler Flow Blockage)
5227	3.1.2	(8)	Loss of component cooling system or cooling to individual components (TBCCW Leakage/Rupture at HX Header)
5228	3.1.2	(8)	Loss of component cooling system or cooling to individual components (Trip of all Drywell Cooling Fans)
5229	3.1.2	(9)	Normal feedwater system failure (Reactor Feed Pump C Trip)
5230	3.1.2	(9)	Normal feedwater system failure (Loss of Extraction Steam to Feedwater Heater 4B)
5231	3.1.2	(10)	Loss of all feedwater (normal and emergency) (RCIC Turbine Trip, HPCI Auto Start Failure, Reactor Feed Pumps A, B, and C Trip)
5232	3.1.2	(11)	Loss of protective system channel (Failure to Scram – Low Power Natural Circulation MSIVs Closed ATWS)
5233	3.1.2	(11)	Loss of protective system channel (Failure to Scram – High Power MSIVs Open ATWS)

Plant Malfunction Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5234	3.1.2	(12)	Loss rod failure including drifting rods and misaligned rods (Control Rod 22-27 Drift In)
5235	3.1.2	(12)	Loss rod failure including drifting rods and misaligned rods (Control Rod 26-35 Drift Out)
5236	3.12	(12)	Loss rod failure including stuck rods, uncoupled rods, and dropped rod (Control Rod 26-35 Stuck and Uncoupled)
5237	3.1.2	(12)	Inability to drive control rods (RMCS Sequence Timer Malfunction)
5238	3.1.2	(14)	Fuel cladding failure resulting in high activity in reactor coolant (Fuel Cladding Failure)
5239	3.1.2	(15)	Turbine trip (Main Trip)
5240	3.1.2	(16)	Generator trip (Main Generator Negative Phase Sequence - Trip Generator Load Reject
5241	3.1.2	(17)	Failure in automatic control system(s) that affect reactivity and core heat removal (Actuation of the Recirc B 30 percent Limiter)
5242	3.1.2	(17)	Failure in automatic control system(s) that affect reactivity and core heat removal (Inadvertent (spurious) HPCI Initiation)
N/A	3.1.2	(18)	Failure of reactor coolant pressure and volume control systems (PWR) (Certification Exception Number: 002)
5243	3.1.2	(19)	Reactor Trip (Control rods in Scram Group No. 1 receive a scram signal – 45 control rods scrammed)
Note	3.1.2	(20)	Main steam line break inside containment (Note: Conducted under Transient Performance Test 5309, maximum size unisolable main steam line rupture)

Plant Malfunction Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
Note	3.1.2	(20)	Main steam line break outside containment (Note: Conducted under Malfunction Performance Test 5201, large reactor coolant breaks outside primary containment)
5244	3.1.2	(20)	Main feed line break inside containment (Feedwater Discharge Line B Leak Inside Containment)
5245	3.12	(20)	Main feed line break outside containment (Feedwater Discharge Line A Leak Outside Containment)
5246	3.1.2	(21)	Nuclear instrumentation failure (SRM Channel A Failure)
5247	3.1.2	(21)	Nuclear instrumentation failure (IRM Channel B Failure)
5248	3.1.2	(21)	Nuclear instrumentation failure (APRM Channel B Failure
5249	3.1.2	(21)	Nuclear instrumentation failure (Rod Block Monitor Channel A Failure)
5250	3.1.2	(22a)	Process instrumentation system failure (Recirc Flow Unit A Failure)
5251	3.1.2	(22b)	Process alarm system failure (Loss of Annunciator 1C651 – AR101)
5252	3.1.2	(22b)	Process alarm system failure (Loss of Annunciator 1C601 – AR 110)
5253	3.1.2	(22b)	Process alarm system failure (Loss of Annunciator OC653 – AR015
5254	3.1.2	(22c)	Process control system failure (Recirc Pump A M/A Station Auto Output Failure – High)

Plant Malfunction Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5255	3.1.2	(22c)	Process control system failure (Feedwater Master Level Controller Failure – High)
5256	3.1.2	(23)	Passive malfunction in emergency feedwater (Loss of Core Spray Division 2 Initiation Logic Power and Recirc Loop B Discharge Line Break
5257	3.12	(23)	Passive malfunction in emergency feedwater (RCIC and HPCI Automatic Start Failure When All Feedwater Pumps Trip)
5258	3.1.2	(24)	Failure of the automatic reactor trip system (Failure to Scram on a Loss of All Feedwater Pumps)
5259	3.1.2	(25)	Reactor pressure control system failure including turbine bypass failure (Loss of Turbine Primary and Backup Speed Signals)
5260	3.1.2	(25)	Reactor pressure control system failure including turbine bypass failure (Loss of 24 VDC Logic Bus Power)
5261	3.1.2	(25)	Reactor pressure control system failure including turbine bypass failure (Turbine Pressure Regulator Oscillation)

Steady-State Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5501	App	B1.1	Steady-State Performance – 30 Percent Power
5502	App	B1.1	Steady-State Performance – 75 Percent Power
5503	App	B1.1	Steady-State Performance – 100 Percent Power and One Hour Stability
6501	App	B1.1	Unit Two Steady-State Performance – 30 Percent Power
6502	App	B1.1	Unit Two Steady-State Performance – 70 Percent Power
6503	App	B1.1	Unit Two Steady-State Performance – 100 Percent Power

Computer Real Time Test

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5610	App	A3.1	Simulator Computer Real Time Test

Transient Performance Tests

TEST	ANSI	SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5301	App	B1.2(1)	Transient Performance – Manual scram
5302	App	B1.2(2)	Transient Performance – Simultaneous trip of all feedwater pumps
5303	App	B1.2(3)	Transient Performance – Simultaneous closure of all Main Steam Isolation Valves
5304	App	B1.2(4)	Transient Performance – Simultaneous trip of all recirculation pumps
5305	App	B1.2(5)	Transient Performance – Single recirculation pump trip
5306	App	B1.2(6)	Transient Performance – Main turbine trip (from maximum power level which will not result in an immediate reactor scram)
5307	App	B1.2(7)	Transient Performance – Maximum power ramp (master recirc controller in manual) down to approximately 75 percent and back up to 100 percent
5308	App	B1.2(8)	Transient Performance – Maximum size reactor coolant system rupture combined with a loss of offsite power
5309	App	B1.2(9)	Transient Performance – Maximum size unisolable steam line rupture
5310	App	B1.2(10)	Transient Performance –Simultaneous closure of all Main Steam Isolation valves combined with single stuck open safety/relief valve (inhibit activation of high pressure ECCS)

Simulator Operating Limits Tests

TEST	ANSI SECTION	TITLE/DESCRIPTION OF PERFORMANCE TEST
5601	Section 4.3	Simulator Operating Limits (Drywell floor dp down and high reactor pressure)
5602	Section 4.3	Simulator Operating Limits (High containment pressure and temperature, external pressure, and differential pressure (UP))
5603	Section 4.3	Simulator Operating Limits (Fuel clad temperature)
5604	Section 4.3	Simulator Operating Limits (250 V, 125 V,. and 24 VDC Buses)

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 03/17/98 Test Number: 5101

Test Title: Plant Startup – Cold to Hot Standby (Including operations at hot standby)

Initial Conditions:

This test was performed from a cold shutdown ready to pull control rods condition.

Summary of Test:

The test verified the neutron response to criticality, the simulated equipment's response during a heatup to hot standby conditions, the ability to perform a plant startup – cold to hot standby (including operations at hot standby) in accordance with plant operating procedures, and the presence of appropriate annunciators.

Termination Criteria:

Reactor power is between seven and ten percent, the Main Turbine has been warmed, and the Reactor Mode Switch is ready to be placed in RUN.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 03/18/98 Test Number: 5102

Test Title: Nuclear Startup from Hot Standby to Rated Power (Including turbine startup and generator synchronization)

Initial Conditions:

This test was performed from a hot standby condition.

Summary of Test:

The test verified the neutron response and simulated equipment's response during a Main Turbine startup and Generator synchronization, the increase in reactor power with control rods and Reactor Recirculation Pumps, the ability to perform a nuclear startup from hot standby to rated power (including Turbine startup and Generator synchronization) in accordance with plant operating procedures, and the presence of appropriate annunciators.

Termination Criteria:

Reactor power is at rated.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/15/00 Test Number: 5103

Test Title: Reactor Trip Followed by Recovery to Rated Power

Initial Conditions:

This test was performed from a 100 percent reactor power initial condition. Depressing the manual scram pushbuttons was performed to initiate the test.

Summary of Test:

The test verified the decrease in reactor power and the ability of the EHC System to utilize the Turbine Control Valves and Bypass Valves to control reactor pressure. After plant conditions were stable, XENON was placed into fast time equivalent to seven h ours and a plant startup to 100 percent power was performed. The test verified the neutron response to criticality and the simulated equipment's response during a Main Turbine startup and Generator synchronization, the increase in reactor power with control rods and Reactor Recirculation Pumps, the ability to perform a plant reactor trip followed by recovery to rated power in accordance with plant operating procedures, and the presence of appropriate annunciators.

Termination Criteria:

Plant has been returned to rated reactor power.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/31/01 Test Number: 5104

Test Title: Startup and Power Operations With Less Than Full Reactor Coolant Flow

Initial Conditions:

This test was performed from a cold shutdown condition with only the "B: Reactor Recirculation Pump running.

Summary of Test:

The test verified the neutron response to criticality, the simulated equipment's response during a heatup to hot standby conditions, the increase in reactor power with control rods and Reactor Recirculation flow with only one pump running, the ability to perform a startup and power operations with less than full reactor coolant flow in accordance with plant operating procedures, and the presence of appropriate annunciators.

Termination Criteria:

Maximum power for single reactor pump operation (80 percent rated Recirc Pump speed and reactor power just below the 70 percent rod line.)

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/12/01 Test Number: 5105

Test Title: Shutdown and Power Operations With Less Than Full Reactor Coolant Flow

Initial Conditions:

This test was performed from 36 percent rated reactor power with the "B" Reactor Recirculation Pump secured.

Summary of Test:

The test verified the decrease in reactor power with the insertion of control rods and the reduction of the "A" Reactor Recirculation Pump speed. After the Main Generator was removed from the grid, a manual scram was inserted to complete the reactor shutdown. The test verified the ability to perform a shutdown and power operations with less than full reactor coolant flow in accordance with plant operating procedures.

Termination Criteria:

Plant conditions are stable after the reactor Mode Switch has been placed to Shutdown.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/03/99 Test Number: 5106

Test Title: Plant Shutdown from Rated Power to Hot Standby (Including Load Changes). Followed by a Plant Cooldown to Cold Shutdown Conditions

Initial Conditions:

This test was performed from a 100 percent reactor power initial condition.

Summary of Test:

The test verified the decrease in reactor power with Reactor Recirculation Pumps and Control Rods and the removal of the Main Generator from the grid. After reactor power was reduced to between seven and ten percent power, a manual scram was inserted to complete the reactor shutdown. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed and a plant cooldown was initiated. The test verified the ability to perform a plant shutdown from rated power to hot standby (including load changes) followed by a plant cooldown to cold shutdown conditions in accordance with plant operating procedures and the presence of appropriate annunciators.

Termination Criteria:

Shutdown Cooling is in service, the reactor vessel is flooded to between 240 inches and 265 inches, and reactor coolant temperature is below 200 °F.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 Test Number: 5107

Test Title: Plant Heat Balance

Initial Conditions:

This test was performed from a 100 percent reactor power initial condition.

Summary of Test:

The test utilized Plant Procedure RE-OTP-002, Core Thermal Power Evaluation (Backup Method). The manual heat balance data was collected and utilized to calculate the core thermal power level. The test verified the manual calculation of Core Thermal Power and the Plant Process Computer (OD-3) calculated Core Thermal Power agreed within two percent power of each other.

Termination Criteria:

Manual heat balance data has been collected and the core power level has been determined.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/30/99 Test Number: 5108

Test Title: Shutdown Margin Determination

Initial Conditions:

The test was performed from a cold shutdown condition. Remote Function RF-JM156003 (Shutdown Margin Demonstration) was inserted to initiate the test.

Summary of Test:

The test utilized the Plant Procedure SR-100-003, Shutdown Margin Demonstration, to perform the Diagonally-Adjacent-Rod Method of verifying shutdown margin. The test verified the neutron response while sequentially pulling two adjacent control rods to prescribed positions which required bypassing one control rod from the RSCS rod pattern logic in accordance with plant operating procedures.

Termination Criteria:

Completion of Plant Procedure SR-100-003, Shutdown Margin Demonstration.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/15/99 Test Number: 5109

Test Title: In-Sequence Critical and Shutdown Margin Demonstration

Initial Conditions:

This test was performed from a cold shutdown condition.

Summary of Test:

The test utilized the Plant Procedure SR-100-008, In-Sequence Critical and Shutdown Margin Demonstration, to perform the In-Sequence Critical. Control rods were withdrawn until a stable positive period was established. Control rods were inserted to reduce reactor power below 10E+3 cps and a Special Log was initiated. Control rods were again withdrawn to a stable positive and control rod positions were recorded.

Termination Criteria:

Completion of Plant Procedure SR-100-008, In-Sequence Critical and Shutdown Margin Demonstration.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/13/01 Test Number: 5110

Test Title: Operator Conducted Surveillance Testing on Safety-Related Equipment or Systems

This test consists of the following plant surveillances performed on the following dates from the following conditions. The simulator" response was verified utilizing the acceptance criteria specified in each surveillance procedure.

Date	Initial Condition	Surv. No.	Procedure Title
07/15/97	18	SO-024-001	Monthly Diesel Generator Operability
07/15/97	18	SO-024-013	Offsite Power Source and Onsite Class1E Test
07/16/97	18	SO-030-001	Monthly Control Room Emergency Outside Air Supply Operability
07/25/97	18	SO-030-A03/B03	Control Structure Chilled Water Flow Verification
07/25/97	18	SO-034-001	Secondary Containment Isolation Damper Timing Test
	18	SO-054-A03/B03	Emergency Service Water Flow Verification
08/15/97	18	SO-070-001	Operational Check of Standby Gas Treatment System

Unit One Only

Date	Initial Condition	Surv. No.	Procedure Title
07/21/97	18	SO-100-007	Daily Operating Surveillance (Attachment G, Recirc Jet Pump Operability)
07/16/97	2	SO-100-011	Reactor Vessel Temperature and Pressure Recording
07/21/97	18	SO-116-003	Quarterly RHRSW System Flow Verification
07/25/97	2	SO-131-001	RWM Operability Demonstration Startup/Following System Failure
07/29/97	18	SO-131-003	RWM Operability Demonstration During Decreasing Power
08/20/97	18	SO-134-001	Secondary Containment Isolation Damper Timing Test
08/20/97	18	SO-149-A02	Quarterly RHR System Flow Verification Division 1
08/20/97	18	SO-149-B02	Quarterly RHR System Flow Verification Division 2

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/29/97 Test Number: 5110

Test Title: Operator Conducted Surveillance Testing on Safety-Related Equipment or Systems

This test consists of the following plant surveillances performed on the following dates from the following conditions. The simulator" response was verified utilizing the acceptance criteria specified in each surveillance procedure.

Date	Initial Condition	Surv. No.	Procedure Title
08/20/97	18	SO-149-A05	Quarterly RHR Loop A Valve Exercising
08/22/97	18	SO-149-B05	Quarterly RHR Loop B Valve Exercising
08/19/97	18	SO-150-002	RCIC Pump Flow Verification
08/20/97	18	SO-150-004	Quarterly RCIC Valve Exercising
08/22/97	2	SO-150-005	24 Month RCIC Flow Verification
08/20/97	18	SO-151-A02	Core Spray Flow Verification Division 1
08/20/97	18	SO-151-B02	Core Spray Flow Verification Division 2
08/20/97	18	SO-152-002	HPCI Flow Verification
08/22/97	18	SO-152-004	Quarterly HPCI Valve Exercising
07/22/97	2	SO-152-005	24 month HPCI Flow Verification
08/25/97	2	SO-152-002	24 month SLC Initiation and Injection Demonstration
08/25/97	2	SO-153-003	24 month SLC System Operability
08/25/97	2	SO-153-004	SLC Flow Verification
08/25/97	18	SO-155-006	ARI Manual Trip Channel Functional Test
07/07/97	2	SO-156-003	Refuel Mode One Rod Interlock Check
07/22/97	2	SO-156-004	Rod Sequence Control System Self-Test
07/22/97	2	SO-156-005	RSCS Rod Withdrawal Block After Initial Rod Withdrawal
07/22/97	53	SO-156-006	RSCS Rod Motion Block During Power Reduction
07/24/97	2	SO-156-007	Control Rod Coupling Check
08/24/97	2	SO-156-001	Manual Scram Control Switch Functional Check
08/25/97	2	SO-158-002	24 Month Mode Switch Shutdown Position Check
08/19/97	2	SO-160-001	LOCA Test of Drywell Area Unit Coolers Fans
08/20/97	2	SO-164-001	Recirculation Pump Discharge Valve and Bypass Valve Operability Test

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/29/97 Test Number: 5110

Test Title: Operator Conducted Surveillance Testing on Safety-Related Equipment or Systems

This test consists of the following plant surveillances. For date of performance and complete listing of surveillances see certification test 5110.

Deficiency S5010147 ESW Quarterly Flow surveillance's SO-054-A02 and SO-054-B02

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/13/99 Test Number: 5201

Test Title: Large Reactor Coolant Breaks Outside Primary Containment (MF-MS183008, "D" Main Steam Line Leak Outside Containment (Turbine Building Pipe Tunnel at 100 Percent Severity)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-MS183008 (D"D" Main Steam Line Leak Outside Containment (Turbine Building Tunnel) at 100 percent severity (24 inch double-ended sheer) was inserted to initiate the test.

Summary of Test:

The test verified the reactor pressure and reactor water level responses, the MSIV closure and subsequent reactor scram, the RCIC and HPCI automatic initiations, the Reactor Recirc Pump trips, and the presence of appropriate Control Room annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

Reactor water level has been restored to above +13 inches with RCIC and/or HPCI.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/24/98 Test Number: 5202

Test Title: Small Reactor Coolant Breaks Outside Primary Containment (MF-MS183008, "D" Main Steam Line Leak Outside Containment (Turbine Building Pipe Tunnel) at Two Percent Severity)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-MS183008 ("D" Main Steam Line Leak Outside Containment (Turbine Building Pipe Tunnel) at two percent severity (double-ended sheer) was inserted to initiate the test.

Summary of Test:

The test verified the reactor pressure and reactor water level responses, the Main Turbine Control Valves and Generator power responses to the main steam line pressure reduction, and the steam flow/feedwater flow mismatch. The test also verified the subsequent MSIV closure, reactor scram, subsequent RPV level recovery, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with RCIC after the MSIVs have closed.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/20/98 **Test Number:** 5203

Test Title: Small Reactor Coolant Breaks Inside Primary Containment (MF-RR164010, Reactor Vessel Bottom Head Drain Line Leakage)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR165010 (Reactor Vessel Bottom Head Drain Line Leakage) at one percent severity (8.64 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the drywell pressure and temperature responses, the HPCI and Diesel Generator automatic initiations, the containment isolations, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed. Drywell coolers were started in slow speed per Emergency Operating Procedure EO-100-103 (Primary Containment Control Bases).

Termination Criteria:

Reactor water level is being maintained between +13 inches with RCIC and HPCI.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/13/99 **Test Number:** 5204

Test Title: Failure of Safety and Relief Valves (MF-MS183011M, Safety Relief Valve "M" Fails Open)

Initial Conditions:

This test was performed from 54 percent reactor power, transient Xenon condition. Malfunction MF-MS18310M (Safety Relief Valve "M" Stuck Open) was inserted to initiate the test.

Summary of Test:

The test verified the feedwater and steam flow mismatch and the reduction of Generator load. A reactor scram was inserted and RCIC was started to assist in recovering reactor water level. When main steam line pressure decreased to approximately 650 psig, Malfunction MS183010M (Safety Relief Valve "M" Stuck Open) was removed to allow the SRV to close and the change in the rate of reactor pressure reduction was verified and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

At least two minutes have elapsed after the SRV was closed.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/24/98 **Test Number:** 5205

Test Title: Loss of Instrument Air (MF-1A118001, Instrument Air Leakage 0 Turbine Building Elevation 646')

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-1A118001 (Instrument Air Leakage – Turbine Building Elevation 646') at 100 percent severity (6,042 scfm = 7.7 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of Instrument Air pressure, the automatic start of the standby Instrument Air Compressor and the presence of appropriate annunciators. The reactor was manually scrammed per Off-Normal Procedure ON-118-001 (Loss of Instrument Air). Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After RCIC and HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed. HPCI was secured after the required RPV water level was obtained.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with RCIC.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/11/98 **Test Number:** 5206

Test Title: Loss of Instrument Air (MF-IA118002, Instrument Air Common Header Rupture)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-IA118002 (Instrument Air Common Header Rupture) at 100 percent severity (59082 scfm = 75.3 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of Instrument Air pressure and the automatic start of the Standby Instrument Air Compressor. The reactor was manually scrammed per Off-Normal Procedure ON-118-001 (Loss of Instrument Air). The test verified the outboard MSIVs failed closed and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-001 were performed.

Termination Criteria:

Outboard MSIVs have failed closed and reactor water level is between +13 to +54 inches.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/11/98 **Test Number:** 5207

Test Title: Loss of Instrument Air (MF-IA118003, Instrument Air Leakage – Turbine Building Elevation 699')

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-IA118003 (Instrument Air Leakage – Turbine Building Elevation 699') at 100 percent severity (6042 scfm = 7.7 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of Instrument Air pressure, the automatic start of the Standby Instrument Air Compressor, the scram pilot air valves beginning to open on low air pressure (which causes a reactor scram on high level in the scram discharge volume), the outboard MSIVs failing closed, and the presence of the appropriate annunciators. HPCI and RCIC were utilized to recover reactor water level. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After the MSIVs failed closed, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed.

Termination Criteria:

Outboard MSIVs have failed closed and reactor pressure is being controlled by SRVs, HPCI, or RCIC.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/14/99 **Test Number:** 5208

Test Title: Loss or Degraded Electrical Power to the Station (MF-DS001002 and MF-EG198001, Sustained Station Blackout With Generator Output Breaker Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-EG198001 (Generator Synch Output Breaker 1R101 Spurious Trip) and MF-DS101002 (Station Blackout) was inserted to initiate the test.

Summary of Test:

The test verified the loss of offsite power, the subsequent MSIV closure, the failure of the Diesel Generators to start, the recovery of reactor water level with HPCI and RCIC, and the change to reactor pressure, drywell temperature, and suppression pool temperature due to the lifting Safety Relief Valves, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After approximately ten minutes, applicable operator actions of EO-100-030 (Unit One Response to Station Blackout) specified to be performed within the first 30 minutes of the station blackout were performed which included plant cooldown and pressure reduction with HPCI and (as necessary) SRVs.

Termination Criteria:

All AC sources of power have been lost for at least 30 minutes and reactor pressure is less than 500 psig.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/15/00 **Test Number:** 5209

Test Title: Loss or Degraded Electrical Power to the Station (MF-DC102003, Loss of 125 VDC Bus 1D612)

Initial Conditions:

The test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-DC102003 (Loss of 125 VDC Bus 1D612) was inserted to initiate the test.

Summary of Test:

The test verified the immediate "B" Reactor Recirc Pump trip and the "A" Reactor Recirc Pump scoop tube lockup, the loss of various functions and instrumentation that were powered from 125 VDC Buses 1D614 and 1D615, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-102-01 (Loss of 125 VDC Bus) were performed which resulted in the trip of the "A" Reactor Recirc Pump. The reactor was manually scrammed and applicable operator actions of Procedure ON-100-101 were performed

Termination Criteria:

Power has been lost for at least five minutes, alternate power sources have been transferred, and RPV water level is being maintained between +13 and +54 inches after the manual reactor scram.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/15/00 **Test Number:** 5210

Test Title: Loss or Degraded Electrical Power to the Station (MF-DC102004, Loss of 125 VDC Bus 1D622)

Initial Conditions:

The test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-DC102004 (Loss of 125 VDC Bus 1D622) was inserted to initiate the test.

Summary of Test:

The test verified the immediate "A" Reactor Recirc Pump trip and the "B" Reactor Recirc Pump scoop tube lockup, the loss of various functions and instrumentation that were powered from 125 VDC Buses 1D624 and 1D624, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-102-001 (Loss of 125 VDC Bus) were performed which resulted in the trip of the "B: Reactor Recirc Pump. The reactor was manually scrammed and applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

Power has been lost for at least five minutes, alternate power sources have been transferred, and RPV water level is being maintained between +13 and +54 inches after the second Recirc Pump trip.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/15/00 **Test Number:** 5211

Test Title: Loss or Degraded Electrical Power to the Station (MF-DC102005, Loss of 125 VDC Bus 1D632)

Initial Conditions:

The test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-DC102005 (Loss of 125 VDC Bus 1D632) was inserted to initiate the test.

Summary of Test:

The test verified the indicated feedwater flow less than indicated steam flow, the runback of both Reactor Recirc Pump, the loss of various functions and instrumentation that were powered from 125 VDC Buses 1D634 and 1D635, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-102-001 (Loss of 125 VDC Bus) were performed.

Termination Criteria:

Power has been lost for at least five minutes and alternate power sources have been transferred.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/15/00 **Test Number:** 5212

Test Title: Loss or Degraded Electrical Power to the Station
(MF-DC102006, Loss of 125 VDC Bus 1D642)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-DC102006 (Loss of Power on Bus 1D642) was inserted to initiate the test.

Summary of Test:

The test verified reactor water level increases the failure of the Main Turbine to trip due to high reactor water level, the Reactor Feedwater Pumps trip on high reactor water level, reactor water level decrease causes a reactor scram, the loss various functions and instrumentation that were powered from 125 VDC Buses 1D644 and 1D645, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-00-101 were performed. After RCIC and HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed. Applicable operator actions of Off-Normal Procedure ON-102-001 (Loss of 125 VDC Bus) were performed.

Termination Criteria:

Power has been lost for at least seven minutes, alternate power has been transferred (as applicable), and reactor water level has been restored above +13 inches by HPCI and RCIC after the reactor scram.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/22/00 **Test Number:** 5213

Test Title: Loss or Degraded Electrical Power to the Station (Remote Function RF-DB157005, Uninterrupted Power Supply 1D240 Inverter Output Breaker OPEN, Loss of 1Y218/1Y219.

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Remote Function RF-DB157005 (Uninterruptible Power Supply 1D240 Inverter Output Breaker) to OPEN was inserted to initiate the test.

Summary of Test:

The test verified the closure of the RWCU inlet isolation valve and subsequent RWCU pump trips, the loss of various functions and instrumentation that were powered from Instrument Buses 1Y218 and 1Y219, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-117-001 (Loss of Instrument Bus) were performed.

Termination Criteria:

Alternate power has been transferred and at least four minutes have elapsed.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5214

Test Title: Loss or Degraded Electrical Power to the Station (Remote Function RF-DB106284, 1B226024 Feeder to 1Y226, OPEN, Loss of AC Instrument Bus 1Y226)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Remote Function RF-DB106284 (power supply breaker to AC Instrument Bus 1Y226) to OPEN was inserted to initiate the test.

Summary of Test:

The test verified the shutdown of the operating Instrument Air Compressors, the Zones 1, II and III "B" Fans trip, the closure of the RWCU outlet isolation valve and subsequent RWCU Pump trips, the loss of various functions and instrumentation that were powered from Instrument Bus 1Y226, and the presence of appropriate annunciators. Prior to 65 psig instrument air header pressure, applicable operator actions of Off-Normal Procedure ON-118-001 (Loss of Instrument Air) were performed which crosstied the Service Air System to the Instrument Air System.

Termination Criteria:

Power has been lost for at least eight minutes and Instrument Air has been cross connected to the Service Air System.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/20/98 **Test Number:** 5215

Test Title: Loss or Degraded Electrical Power to the Station
(MF-DS003003, Startup Bus 10 Lockout)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-DS00303 (Startup Bus 10 Lockout) was inserted to initiate the test.

Summary of Test:

The test verified the voltage transient from 4160 VAC Bus power swap resulting in the loss of "A" RPS power, the half reactor scram, the inboard containment isolation, the closure of the RWCU outlet isolation valve and subsequent RWCU Pump trips, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-003-001 (Loss of Startup Bus 10) were performed, including the restart of power from the "A" RPS MG Set.

Termination Criteria:

RPS "A" power has been restored and Startup Bus 10 power has been lost for at least five minutes.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/31/98 **Test Number:** 5216

Test Title: Loss or Degraded Electrical Power to the Station (MF-DS003003, MF-DS003004 and MF-EG198001, SU Buses 10 and 20 Lockout), and MF-EG198001 (Generator Synch Output Breaker 1R101 Spurious Trip) were inserted to initiate the test.

Initial Conditions:

This test was performed from a 38 percent reactor power condition. Malfunctions MF-DS003003 (Startup Bus 10 Lockout), MF-DS003004 (Startup Bus 20 Lockout), and MF-EG198001 (Generator Synch Output Breaker 1R101 Spurious Trip) were inserted to initiate the test.

Summary of Test:

The test verified the MSIV closure and subsequent reactor scram due to the loss of turbine generator and offsite power, the automatic start of RCIC and HPCI, the lifting of Safety Relief Valves, the Drywell temperature increase, the Suppression Pool temperatures increase, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-104-001 (Unit Response of Loss of Offsite Power) were performed, including the recovery of the RPS MG Sets and restoration of RBCCW and TBCCW Systems. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After the first SRV opened on high reactor pressure, the applicable operator actions of Emergency Operating Procedure EO-100-012 (RPV Control Bases) were performed.

Termination Criteria:

Offsite power has been lost for at least 15 minutes.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/21/98 **Test Number:** 5217

Test Title: Loss or Degraded Electrical Power to the Station (MF-DS103005A, Auxiliary Bus 11A (1A101) Fails to Transfer)

Initial Conditions:

This test was performed from a 20 percent reactor power condition. Malfunctions MF-DS103005A (Auxiliary Bus 11A Fails to Fast Transfer) and MF-TC193001 (Turbine Trip) were inserted to initiate the test.

Summary of Test:

The test verified the trip of the “A” Reactor Recirc Pump, the reduction of core flow and reactor power, the trip of the “A” and “C” Condensate Pumps and Circulating Water Pumps, the trip of “A” Service Water Pump, the runback of the “B” Reactor Recirc Pump, the loss of various load centers, and the presence of appropriate annunciators.

Termination Criteria:

At least four minutes have elapsed and reactor water level has been restored to normal level using the Condensate and Feedwater Systems.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/21/98 **Test Number:** 5218

Test Title: Loss or Degraded Electrical Power to the Station (MF-DS104001B and MF-DG024001B, 4.16 kV Bus 1B Differential Current Lockout Trip and Diesel Generator "B" Failure)

Initial Conditions:

The test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-DG024001B (Diesel Generator "B" Fails to Start) and MF-DS104001B (4.16 kV ESS Bus 1B Differential Current Lockout Trip) were inserted to initiate test.

Summary of Test:

The test verified the simulator's response to a loss of one 4 kV Bus, which caused various pumps to be de-energized, the shutdown of the operating Instrument Air Compressors, a half scram and PCIS isolation, the trip and automatic start of various HVAC Fans, and the presence of appropriate annunciators.

After instrument air pressure decreased, both malfunctions were removed and protective relays were reset, which allowed the Diesel to start and pick up loads on the 4 kV Bus. The Instrument Air Compressors were manually restarted.

Termination Criteria:

Power has been restored to the 1B 4 kV Bus and Instrument Air pressure is above 90 psig.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/21/98 **Test Number:** 5219

Test Title: Loss or Degraded Electrical Power to the Station (MF-DS103002, 13.8 kV Bus 11B Lockout)

Initial Conditions:

This test was performed from 80 percent reactor power condition. Malfunction MF-DS103002 (13.8 kV Bus 11B Lockout) was inserted to initiate the test.

Summary of Test:

The test verified the trip of the "B" Reactor Recirc Pump, the "B" and "D": Condensate Pumps, the "B" and "D" Circulating Water Pumps, and the "B": Service Water Pump, the runback of the "A" Reactor Recirc Pump, the automatic transfer of power source to the Standby Service Water Pump, the automatic start of the Standby Service Water Pump, the core flow and reactor power reduction, the reactor water level swell due to the Reactor Recirc Pump trip, and the presence of appropriate annunciators.

Termination Criteria:

At least three minutes have elapsed and reactor water level has been restored to normal level by the Feedwater System.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/16/99 **Test Number:** 5220

Test Title: Loss of Condenser Vacuum Including Loss of Condenser Level Control (MF-MS143007D, Main Condenser Tube Leaks at One Percent Severity)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-MC143007D (LP Waterbox Leak) at one percent severity (0.25 tube, 13 lbm/sec) was inserted to initiate test.

Summary of Test:

The test verified the Condensate Pump discharge conductivity, Feedwater Header conductivity, RWCU inlet conductivity increase, and the presence of appropriate annunciators. After Condensate Pump discharge conductivity reached at least 5 umho/cm, the reactor power was reduced using Reactor Recirc Pumps. The Condensate Pump discharge conductivity was verified to increase at a greater rate after the power reduction.

Termination Criteria:

Condensate Pump discharge conductivity has increased to at least 7 umho/cm.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/28/99 **Test Number:** 5221

Test Title: Loss of Condenser Vacuum Including Loss of Condenser Level Control (Component Level Failure LIC-1051A and LIC-10514B Condenser Water Level Controllers)

Initial Conditions:

This test was performed from approximately 25 percent reactor power. Component Level Failure or LIC-10514A Auto Output to 100 percent (maximum output signal) and Component Level Failure of LIC-10514B Auto Output to 0 percent (minimum output signal) were inserted to initiate the test.

Summary of Test:

The test verified the increase of condensate makeup flow to the Main Condenser, reduction of condensate reject flow from the Main Condenser, increase of Main Condenser hotwell level, decrease of Main Condenser vacuum, slight reduction of Main Generator load, trip of the Main Turbine on low vacuum which resulted in a reactor scram and trip of both Reactor Recirculation Pumps, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

At lease five minutes have elapsed since the main turbine trip and reactor water level have been restored between +13 and +54 inches with the Feedwater System.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/28/99 **Test Number:** 5222

Test Title: Loss of Condenser Vacuum (Circ Water Pumps A, B, C, and D Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Circulating Water Pumps A, B, C, and D were tripped using component level failures (pump breaker mechanical operation – open) to initiate the test.

Summary of Test:

The test verified the response to the manual reduction of reactor power with Reactor Recirculation Pumps, the Main Generator load reduction, the eventual trip of the Main Turbine on low condenser vacuum, the reactor scram, the resultant reactor pressure and reactor water level responses, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

At least 15 minutes have elapsed since the loss of main condenser vacuum.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/07/99 **Test Number:** 5223

Test Title: Loss of Service Water (MF-SW111002 A, B, and C Service Water Pump Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. After winter conditions were established, Malfunction MF-SW111002A (Service Water Pump "A" Trip) was inserted to initiate the test.

Summary of Test:

The first part of the test verified the Standby Service Water Pump did not automatically start, nor did any served system overheat to an alarm state. The second part of the test verified the simulator's response to a second Service Water Pump trip where the Standby Service Water Pump automatically started on low header pressure and the presence of appropriate annunciators. After five minutes had elapsed, the applicable operator actions of Off-Normal Procedure ON-111-001 (Loss of Service Water) were performed where reactor power was reduced using Recir Pumps and Control Rods and TBCCW, RBCCW, and Control Structure Chiller loads were supplied by the Emergency Service Water System. The third part of the test was performed after a total of 20 minutes had elapsed. This portion of the test verified the simulator's response to a loss of the third and final Service Water Pump, which resulted in the trip of the Main Generator and subsequent reactor scram. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

At least seven minutes have elapsed since the last Service Water Pump was tripped and reactor water level is being maintained above +13 inches after the turbine trip and reactor scram.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/29/98 **Test Number:** 5224

Test Title: Loss of Shutdown Cooling (MF-EW-116001, RHRSW Loop 1A Leak/Rupture)

Initial Conditions:

This test was performed from a cold shutdown, EOL condition with Shutdown Cooling in service. Malfunction MF-EW116001 (RHRSW Loop 1A Leak/Rupture) at 100 percent severity (9,756 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the decrease of RHR Service Water Pump Discharge pressure, the increase of RHR Service Water flow into the break, the increase in RHRSW Pump current, the RHR heat exchanger discharge line temperature increase, and the presence of appropriate annunciators. Sixty seconds later, the RHRSW Pump was manually tripped.

Termination Criteria:

RHR heat exchanger discharge line temperature has increased at least five degrees.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/11/01 **Test Number:** 5225

Test Title: Loss of Component Cooling System or Cooling to Individual Components (MF-TW115002A, CRD Pump "A" Bearing and Lube Oil Cooler Flow Blockage)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-TW115002A (CRD Pump "A" Bearing and LO Cooler Flow Blockage) at 100 percent severity (100 percent blockage) was inserted to initiate the test.

Summary of Test:

The test verified the increase of CRD Pump "A" stator temperature and current, the eventual CRD Pump trip, the CRD system flow and pressure decay, subsequent presence of control rod scram accumulator trouble lights, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Operating Procedure ON-155-007 (Loss of CRD System Flow) were performed, which resulted in the "B" CRD Pump start and the return of the CRD System to normal.

Termination Criteria:

CRD system flow has been restored to normal with the Standby CRD Pump and all the CRD accumulator lights have cleared.

TESTS ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/11/01 **Test Number:** 5226

Test Title: Loss of Component Cooling System or Cooling to Individual Components (MF-RR164020A, Reactor Recirc "A" M/G Set Hydraulic Fluid Cooler Flow Blockage)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR164020A (Reactor Recirculation "A": M/G Set Hydraulic Fluid Cooler Flow Blockage) at 100 percent severity (100 percent blockage) was inserted to initiate.

Summary of Test:

The test verified the "A" Reactor Recirculation M/G Set component temperatures increase, the resultant M/G Set and Recirc Pump trip, the core flow and reactor power reduction, the resultant reactor water level response, and the presence of appropriate annunciators.

Termination Criteria:

At least two minutes have elapsed after the "A" Reactor Recirc Pump trip.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/11/01 **Test Number:** 5227

Test Title: Loss of Component Cooling System or Cooling to Individual Components (MF-TW115001, TBCCW Leakage/Rupture at HX Header)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-TW115001 (TBVVW Leakage/Rupture at HX Header) at 100 percent severity (1,038 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of TBCCW pressure, the start of Standby TBCCW Pump, the trip of the Instrument Air and Service Air Compressors, the Condensate Pumps bearing Temperature increase, and the presence of appropriate annunciators. Applicable operator action (manually scramming the reactor) of Off-Normal Procedure ON-118-001 (Loss of Instrument Air) was performed. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

Instrument Air and Service Air Compressors have tripped, reactor water level is being maintained between +13 and +54 inches with the Reactor Feedwater System, and at least ten minutes.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/31/01 **Test Number:** 5228

Test Title: Loss of Component Cooling System or Cooling to Individual Components (MF-PC159005A-R, Trip of All Drywell Cooling Fans)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF- PC159005A-R (Drywell Cooling Fans 1V411A through 1V418B trip) were inserted to initiate the test.

Summary of Test:

The test verified the resultant containment atmosphere temperature and pressure increase and the presence of appropriate annunciators. The drywell was vented per Operating Procedure OP-173-001 (Containment Atmosphere Control System).

Termination Criteria:

Drywell pressure has increased to greater than 1.72 psig while venting through the SBGT System and Reactor Scrams.

PREPARED BY: F. Tarselli Date: 09/14/01

APPROVED BY: W. W. Hunt Date: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/29/98 **Test Number:** 5229

Test Title: Normal Feedwater System Failure (MF-FW145009C, Reactor Feedwater Pump "C" Trip)

Initial Conditions:

This test was performed from 100 percent reactor power equilibrium Xenon condition. Malfunction MF-FW145009C (Reactor Feed Pump "C" Trip) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of Instrument Air pressure, the automatic start of the standby Instrument Air Compressor and the presence of appropriate annunciators. The reactor was manually scrammed per Off-Normal Procedure ON-118-001 (Loss of Instrument Air). Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After RCIC and HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed. HPCI was secured after the required RPV water level was obtained.

Termination Criteria:

Reactor power and water level has stabilized and a total of at least five minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 07/31/98 **Test Number:** 5230

Test Title: Normal Feedwater System Failure (Loss of Extraction Steam to Feedwater Heater 4B)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Component Level Fail CLOSED LV-10404B (Feedwater Heater 4B Dump Level Control Valve) and LV-10408B (Feedwater Heater 4B Level Control Valve) were inserted to initiate the test.

Summary of Test:

The test verified the Feedwater Heater 4B water level increase, the failure of the emergency dump valve to open, the appropriate feedwater and extraction steam responses as the 4B Heater isolates on high level, the reactor power increase due to the feedwater temperature decrease, and the presence of the appropriate annunciators. After reactor power stabilized, the applicable operator actions of Off-Normal Procedure ON-147-001 (Loss of Feedwater Heating – Extraction Steam) were performed, which included reducing reactor power with Recirc flow and the isolation of appropriate extraction valves.

Termination Criteria:

Reactor power has been reduced by 20 percent, and feedwater temperatures have stabilized.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/29/98 **Test Number:** 5231

Test Title: Loss of All Feedwater (Normal and Emergency) (MF-RC150011, MF-HP152002, MF-FW145009A, B, C RCIC Trip, HPCI Auto Start Failure, RX Feedwater Pump Trips)

Initial Conditions:

This test was performed from a 100 percent power equilibrium Xenon condition. Malfunctions MF-RC150011 (RCIC Trip), MF-HP152002 (HPCI Auto Start Failure), and MF-FW145009A, B, C (Reactor Feedwater Pumps A, B, C Trip) were inserted to initiate the test.

Summary of Test:

The test verified the generator power reduction, the reactor scram on low reactor water level. The inability of HPCI, RCIC and Feedwater Pumps to recover reactor water level, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After the Reactor Recirc Pumps tripped on low vessel level, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed, which included the reduction of reactor pressure using the bypass valves and the recovery of reactor water level with the Condensate System.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with the Feedwater System.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/12/98 **Test Number:** 5232

Test Title: Loss of Protective System Channel (MF-RP158003, Failure to Scram – Low Power Natural Circulation MSIVs Closed ATWS)

Initial Conditions:

This test was started from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-RP158003 (Failure to Scram) and MF-RR164019A, B (Recirculation Pumps "A" and "B" Trip) were inserted to establish a natural circulation condition with the reactor at power. Alternate Rod Insertion (ARI) System was prevented from functioning. Malfunction MF-MS183002 (Outboard MSIVs Fail Closed) was inserted to initiate the test.

Summary of Test:

The test verified the simulator's response to an ATWS from natural circulation condition. RCIC and HPCI were allowed to automatically initiate to establish conditions that would match reference data. Stabilization of reactor water level and reactor power with HPCI, RCIC, and CRD injecting into the reactor was verified. Standby Liquid was prevented from injecting into the reactor for ten minutes. Applicable operator actions of Procedure ON-100-101 were performed. Applicable operator actions of Emergency Operating Procedure EO-100-113 (Level/Power Control Bases) were performed. After ten minutes, Standby Liquid Control System was allowed to successfully start and inject into the reactor vessel.

Termination Criteria:

All APRMs indicate less than one percent power after SLC injection into RPV.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/13/98 **Test Number:** 5233

Test Title: Loss of Protection System Channel (MF-RP158003, Failure to Scram – High Power MSIVs Open ATWS)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Alternate Rod Insertion (ARI) System was prevented from functioning. Malfunctions MF-RP158003 (Failure to Scram) and MF-NM178007C, D (APRM "C" and "D" Channel Failure) at 100 percent severity (125 percent indicated flux) were inserted to initiate test.

Summary of Test:

The test verified the failure to scram at high power. Applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control) were performed, including reduction of reactor power by running back Recirc Pumps speed and tripping the Recirc Pumps. Applicable operator actions of Emergency Operating Procedure EO-100-113 (Level/Power Control Bases) were performed, including inhibiting HPCI and RCIC injection and the reduction of reactor water level with the Feedwater System. After reactor water level was below -80 inches, applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed, including insertion of control rods by maximizing CRD system flow to drift controls into the core.

Termination Criteria:

Reactor power has been reduced to below one percent by lowering reactor water level and by drifting control rods in with CRD maximized.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/22/99 **Test Number:** 5234

Test Title: Control Rod Failure Including Drifting and Misaligned Control Rods (MF-RD155004, Control Rod 22-27 Drift In)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RD155004 (Control Rod 22-27 Drift In) at 10 percent severity (4.8 inches per second) was inserted to initiate the test.

Summary of Test:

The test verified the indication of a drifting control rod on the Full Core Display, the Power Plex printer, and the Alarm CRT, the reduction of generator MWe, total core power, and the LPRM readings around the drifting control, and the presence of appropriate annunciators.

Termination Criteria:

Reactor power is stable after Control Rod 22-27 has drifted to the Full-In Overtravel position.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/33/99 **Test Number:** 5235

Test Title: Control Rod Failure Including Drifting and Misaligned Control Rods (MF-RD155005, Control Rod 26-35 Drift Out)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RD155005 (Control Rod 26-35 Drift Out) was inserted to initiate the test.

Summary of Test:

The test verified the indication of a drifting control rod on the Full Core Display, the Power Plex printer, and the alarm CRT, the increase of generator MWe, total core power, and the LPRM readings around the drifting control, and the presence of appropriate annunciators.

Termination Criteria:

Reactor power has stabilized after Control Rod 26-35 has drifted to the Full Out position.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 09/08/99 **Test Number:** 5236

Test Title: Control Rod Failure Including Stuck Rod, Uncoupled Rod, and Dropped Rod (MF-RD155006 and MF-RD155007, Control Rod 26-35 Stuck and Uncoupled)

Initial Conditions:

This test was performed from a 32 percent reactor power startup Xenon condition. Malfunction MF-RD155006 (Control Rod 26-35 Stuck) was inserted and Control Rod 26-35 withdrawal was attempted to initiate the test.

Summary of Test:

The test verified the inability to manually move the stuck control rod with normal and elevated drive water system pressures. After the control rod was verified stuck at a fully inserted position, Malfunction MF-RD155007 (Control Rod 26-35 Uncoupled) was inserted. The Control Rod 26-35 mechanism was withdrawn to an overtravel position. Verification of LPRM and Generator MWe changes indicating control rod movement had not occurred and the presence of appropriate annunciators were performed. Malfunction MF-RD155006 (Control Rod 26-35 Stuck) was removed to allow the control rod to drop. LPRM and Generator MWe changes indicating a control rod drop had occurred were verified.

Termination Criteria:

Reactor power has stabilized at a new higher power after the control rod drop.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/10/99 **Test Number:** 5237

Test Title: Inability to Drive Control Rods (MF-LC156001, RMCS Sequence Timer Malfunction)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. A continuous withdrawal of a fully inserted control rod was started followed by the insertion of Malfunction MF-LC156001 (RMCS Sequence Timer Malfunction) to initiate the test.

Summary of Test:

The test verified simulator's response to the attempted continuous withdrawal of a control rod, the initial normal rod motion indication outward, including the increase of reactor power, the interruption of the rod motion and reactor power increase, the settling of the rod at a position other than fully withdrawn, the inability to manually move the control rod, and the presence of appropriate annunciators.

Termination Criteria:

Control rod motion has stopped, and the control rod cannot be moved in either direction.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/13/99 **Test Number:** 5238

Test Title: Fuel Cladding Failure Resulting in High Activity in Reactor Coolant (MF-RR179003, Fuel Cladding Failure)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR179001 (Fuel Cladding Failure) at five percent severity (50 Fuel Rods Fail) at a ramp rate of 30 seconds was inserted to initiate the test.

Summary of Test:

The test verified the increase of Main Steam Line radiation levels. Reactor Recirculation pump speeds were manually reduced to reduce reactor power.

The severity of Malfunction MF-RR179001 (Fuel Cladding Failure) was increased to 100 percent severity (1,000 Fuel Rods Fail) at a five minute ramp rate. The test verified the resultant reactor scram on high Main Steam Line radiation, the increase in radiation levels both inside and outside Primary Containment, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. When the Reactor Feedwater Pumps were no longer able to inject into the reactor vessel, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed.

Termination Criteria:

MSIVs have closed on high MSL radiation, reactor water level is being maintained by RCIC and/or HPCI, and the suppression pool radiation levels have increased as a result of the SRV actuation.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/13/98 **Test Number:** 5239

Test Title: Turbine Trip (MF-TC193001, Main Turbine Trip)

Initial Conditions:

This test was performed from approximately 64 percent reactor power startup Xenon condition. Malfunction MF-TC193001 (Main Turbine Trip) was inserted to initiate the test.

Summary of Test:

The test verified the reactor scrammed on the Main Turbine Trip, the Main Turbine Valve Closures, the Bypass Valves opening to control reactor pressure, the Reactor Recirculation EOC-RPT breaker trips, the reactor vessel water level and reactor pressure responses, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

At least four minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 09/04/98 **Test Number:** 5240

Test Title: Generator Trip (MF-EG198008, Negative Phase Sequence Trip – Generator Load Reject)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-EG198008 (Negative Phase Sequence Trip) was inserted to initiate the test.

Summary of Test:

The test verified the reactor scrammed on the Main Turbine trip, the Main Turbine Valve closures, the Bypass Valves and Safety Relief Valves opening to control reactor pressure, the Reactor Recirculation EOC-RPT breaker trips, the reactor vessel water level and reactor pressure responses, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

Reactor water level has stabilized, and at least four minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/10/99 **Test Number:** 5241

Test Title: Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal (MF-RR164016B, Actuation of the Recirc "B" 30 Percent Limiter)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR164016 (Actuation of the Recirc "B" 30 Percent Limiter) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of Reactor Recirculation Pump "B" speed, reduction of reactor core flow and reactor power, the reactor water level swell, and the presence of appropriate annunciators. Applicable operator actions of Off-Normal Procedure ON-164-002 (Loss of Recirculation Flow) were performed, which include the reduction of the "A" Recirculation Pump speed and the insertion of control rods to below the 80 percent rod line.

Termination Criteria:

Reactor power has been reduced to less than the 70 percent rod line.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/13/99 **Test Number:** 5242

Test Title: Failure in Automatic Control System That Affects Reactivity and Core Heat Removal (MF-HP152004, Inadvertent (Spurious) HPCI Initiation)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-HP152004 (Inadvertent – Spurious HPCI Initiation) was inserted to initiate the test.

Summary of Test:

The test verified the startup and injection of HPCI into the RPV, the increase of reactor power and Generator MWe, and the presence of appropriate annunciators. After reactor power stabilized, applicable operator actions of Off-Normal Procedure ON-156-001 (Unexplained Reactivity Change) were performed, which reduced reactor power below 80 percent by reducing Recirculation Pump speeds. HPCI flow was later reduced and the HPCI System was removed from service per Emergency Support Procedure (ES-149-101) (Overriding ECCS Pump Initiation).

Termination Criteria:

HPCI has been shut down, and reactor power is stable.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 08/31/01 **Test Number:** 5243

Test Title: Reactor Trip (1Y201AF18A and LIS-B21-NO24D, Control Rods in Scram Group No. 1 Receive a Scram Signal – 45 Control Rods Scrammed)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Component Level Fail Reactor Recirc LIS-B21-NO24D1 (NSSSS L3 DIV 2 – 1C005) "SETPOINT DRIFT" to 60 inches on Reactor Protection Fuse 1Y201AF18A (RPS TRIP SYS A PILOT SCRAM VLV SOLENOID GROUP 1 AND LIGHTS) to "OPEN" to initiate the test.

Summary of Test:

The test verified the insertion by scram of 45 control rods, the reactor power reduction caused by the partial reactor scram, the reactor water level decrease, and the presence of appropriate annunciators. The test also verified reactor water level continued to decrease low enough to cause a full automatic reactor scram, the absence of SRV lifts, and the recovery of reactor water level by the Feedwater System.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with the Reactor Feedwater System.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/31/01 **Test Number:** 5244

Test Title: Main Feed Line Break Inside Containment (MF-FW145002, Feedwater Discharge Line "B" Leak/Rupture Inside Containment)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-FW145002 (Feedwater Discharge Line Leak Inside Containment) at 100 percent severity (0.847 square feet or 10,650 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test (an unisolable feedwater line break inside the containment) verified the reactor scram on high drywell pressure, the rapid decrease of reactor pressure and resultant swell in indicated reactor water level, the indications of HPCI and Feedwater Systems discharging into the break, the elevated primary containment temperatures and pressures, the eventual injection of low pressure Emergency Core Cooling Systems, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed.

Termination Criteria:

Reactor water level has been returned to above +13 inches and the LPCI and Core Spray Systems and a total of at least 20 minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/13/98 **Test Number:** 5245

Test Title: Main Feed Line Break Outside Containment (MF-FW145001, Feedwater Discharge Line "A" Leak Outside Containment)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-FW145001 (Feedwater discharge line leak outside containment) at 100 percent severity (57,020 lbm/sec) was inserted to initiate the test.

Summary of Test:

The test verified the rapid reactor water level reduction due to the loss of actual feedwater flow to the RPV, the automatic closure of MSIVs on high steam tunnel temperature, the runback of Reactor Recirculation Pumps after the reactor scram, the start of HPCI and RCIC and trip of Reactor Recirculation Pumps on low reactor water level, the indications of RCIC and Feedwater Systems discharging into the break, and the presence of appropriate annunciators. After reactor water level was restored to above +13 inches, the RCIC System was manually shut down and the ruptured Feedwater Line was isolated.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with HPCI, and the Feedwater break has been isolated.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/04/01 **Test Number:** 5246

Test Title: Nuclear Instrumentation Failure (MF-NM178002A, SRM Channel "A" Failure)

Initial Conditions:

This test was performed from a ready to pull control rods from cold shutdown condition. Malfunction MF-NM178002A (SRM Channel "A" Failure) at 100 percent severity (maximum output, 10E+6 cps) was inserted to initiate the test.

Summary of Test:

The test verified the full-scale indication of SRM "A" countrate, the subsequent SRM "A" positive period transient, the activation of a control rod withdrawal block, and the presence of appropriate annunciators. After manually bypassing SRM "A", the rod withdrawal block and various annunciators were verified to clear.

Termination Criteria:

SRM Channel "A" has been bypassed, and all rod blocks have been cleared.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/30/01 **Test Number:** 5247

Test Title: Nuclear Instrumentation Failure (MF-NM178004B, IRM Channel "B" Failure)

Initial Conditions:

This test was performed from a Ready-To-Go-To-Run, 955 psig reactor pressure condition. Malfunction MF-NM178004B (IRM Channel "B" failure) at 100 percent severity (maximum output, 125 percent flux) was inserted to initiate the test.

Summary of Test:

The test verified the full-scale indication of IRM "B" flux, the activation of a control rod withdrawal block and RPS "B" half scram, and the presence of appropriate annunciators. After manually bypassing IRM "B", the rod withdrawal block and various annunciators were verified to clear. The ability to reset the RPS "B" half scram was verified.

Termination Criteria:

IRM Channel "B" has been bypassed, and the rod block and half scram have been cleared.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 07/30/01 **Test Number:** 5248

Test Title: Nuclear Instrumentation Failure (MF-NM178007C, APRM Channel "C" Failure)

Initial Conditions:

This test was performed from a 100 percent reactor equilibrium Xenon condition. IRM "C" was manually inserted into the RPV. After the upscale indication of IRM "C" was verified. Malfunction MF-NM178007C (APRM Channel "C" failure) at 0 percent severity (minimum output, 0 percent power) was inserted to initiate the test.

Summary of Test:

The test verified the downscale indication of APRM "C" flux, the APRM downscale – companion IRM upscale reactor half scram, the control rod withdrawal block, and the presence of appropriate annunciators. After manually bypassing IRM "C" and APRM "C", the rod withdrawal block and various annunciators were verified to clear. The ability to reset the RPS "A" half scram was verified.

Termination Criteria:

IRM Channel "C" and APRM Channel "C" have been bypassed, and the rod block and half scram have been cleared.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/20/01 **Test Number:** 5249

Test Title: Nuclear Instrumentation Failure (MF-NM178009A, Rod Block Monitor Channel "A" Failure)

Initial Conditions:

This test was performed from a 100 percent reactor equilibrium Xenon condition. Malfunction MF-NM178009A (Rod Block Monitor Channel A Failure) at 109 percent severity (109/125 scale) was inserted to initiate the test.

Summary of Test:

The test verified the Rod Block Monitor initiated a rod withdrawn block, control rods could be inserted but not withdrawn, the ability to bypass the Rod Block Monitor control rod block and withdrawn an inserted control rod, and the presence of appropriate annunciators. After manually bypassing the RBM Channel "A", the rod withdrawal block and various annunciators were verified to clear.

Termination Criteria:

RBM Channel "A" has been bypassed and an inserted control rod can be withdrawn.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5250

Test Title: Process Instrumentation System Failure (MF-NM178012A, Recirc Flow Unit "A" Failure)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-NM178012A (Recirc Flow Unit "A" Failure) at 0 percent severity (minimum output, 0 percent flow) was inserted to initiate the test.

Summary of Test:

The test verified the reduction of the "A" Recirc flow recorder indication, the "A" RBM trip causing a rod withdrawal block, the A, C, E APRMs upscale trip causing a half reactor scram, and the presence of appropriate annunciators. After manually bypassing the "A" Recirc Flow Unit and placing the Recirc Flow Unit "A" Mode Switch to ZERO utilizing Remote Function RF-NM178006, the rod withdrawal block and various annunciators were verified to clear. The ability to reset the RPS "A" half scram was verified.

Termination Criteria:

Recirc Flow Unit A has been bypassed and the rod block and half scram signals have been cleared.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/11/00 **Test Number:** 5251

Test Title: Process Alarm System Failure (MF-AN191001, Loss of Annunciator 1C651 – AR101)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-AR191001 (Loss of Annunciator 1C651 – AR101) was inserted to initiate the test.

Summary of Test:

The test verified the absence of AR101 annunciator alarms on Panel 1C651, while the RWCU Return Isolation Valve (F042) was closed, the automatic trip of both RWCU Pumps, and the presence of other appropriate annunciators.

Termination Criteria:

RWCU Pump has tripped and there are no AR101 (A01-A07 B01-B07, C01-H06) alarms on Panel 1C651.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5252

Test Title: Process Alarm System Failure (MF-AN191006, Loss of Annunciator 1C601 – AR110)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-AR191006 (Loss of Annunciator 1C601 – AR110) and MF-MS183011M (Safety Relief Valve "M" inadvertent opening) were inserted to initiate the test.

Summary of Test:

The test verified the absence of AR110 annunciator alarms on Panel 1C601, while the "M" SRV opened and remained open, the Main Steam flow/Feedwater flow mismatch, the Generator load reduction, the reactor pressure reduction, the presence of the "MR SRV Acoustic Monitor Light, and the presence of the other appropriate annunciators.

Termination Criteria:

The "M" SRV has opened, generator load has stabilized, and there are no AR110 alarms on Panel 1C601.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5253

Test Title: Process Alarm System Failure (MF-AN191010, Loss of Annunciator OC653 – AR015)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-AR191010 (Loss of Annunciator OC653 – AR015), MF-DG024001B (Diesel Generator "B" Fails to Start), and MF-DS104001B (4.16 kV Bus 1B Differential Current Lockout Trip) were inserted to initiate the test.

Summary of Test:

The test verified the absence of AR015 annunciator alarms on Panel OC653, while the 4.16 kV Bus 1B de-energized, the loss of power to various equipment, the activation of a half reactor scram, the activation of containment isolations from the loss of the "B" RPS Bus, and the presence of other appropriate annunciators.

Termination Criteria:

4160 VAC "B" Bus has been de-energized for at least eight minutes.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5254

Test Title: Process Control System Failure (MF-RR164008A, Recirc Pump "A" M/A Station Auto Output Failure – High)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR164008A (Recirc Pump "A" M/A Station Auto Output Failure) at 100 percent severity (maximum output) was inserted to initiate the test.

Summary of Test:

The test verified the "A" Reactor Recirculation Pump speed, total core flow, Turbine Generator output, and reactor power increases. The "A" Reactor Recirculation Pump speed was reduced to its initial value by taking manual control of the Recirc Pump "A" M/A controller.

Termination Criteria:

Reactor power and core flow has been returned to their initial values by manual control of the "A" Reactor Recirc Pump.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5255

Test Title: Process Control System Failure (Feedwater Master Level Controller Failure – High)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Component Level fail FW LEVEL CTL/DEMAND SIGNAL (LIC-C32-1R600) AUTO OUTPUT to 100 percent output was inserted to initiate the test.

Summary of Test:

The test verified the increase of reactor water level and reactor power followed by a Main Turbine and RFP trips, the reactor scram, the Turbine Valve closures, the Bypass Valves opening to control reactor pressure, the Reactor Recirculation EOC-RPT breaker trips, and the automatic initiation of RCIC on low reactor water level. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. Applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed, which included restarting a Reactor Feedwater Pump and removing the RCIC from service.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with the Reactor Feedwater System after RCIC has been manually shut down.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/19/00 **Test Number:** 5256

Test Title: Passive Malfunction in Emergency Feedwater (MF-CS151005 and MF-RR164011B, Loss of Core Spray Division 2 Initiation Logic Power and Recirc Loop "B" Line Break)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-CS151005 (Loss of Core Spray Division 2 Initiation Logic Power) and RR164011B (Recirc Loop "B" Discharge Line Break) at a severity of 100 percent (4.17 square feet) was inserted to initiate the test.

Summary of Test:

The test verified the reactor scram, reduction of reactor water level and pressure, the increase of containment temperature and pressure, the failure of the "B" and "D" Core Spray Pumps and the "B" Diesel Generator to automatically start during the LOCA, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After the "B" Diesel Generator was manually started, the "B" and "D" Core Spray Pumps were manually started per Emergency Operating Procedure EO-100-102 (RPV Control Bases).

Termination Criteria:

Reactor water level has been raised to above +13 inches and at least ten minutes have elapsed.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/20/00 **Test Number:** 5257

Test Title: Passive Malfunction in Emergency Feedwater (MF-RC150001, MF-HP152002, MF-FW145009A/B/C, RCIC and HPCI Auto Start Failures When All FW Pumps Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-RC150001 (RCIC Automatic Start Failure), MF-HP152001 (HPCI Automatic Start Failure), and MF-FW145009A,B,C (Reactor Feedwater Pumps A, B, and C Trip) were inserted to initiate the test.

Summary of Test:

The test verified the simulator's initial response to a loss of all feedwater (normal and emergency), the reactor scram, the trip of both Reactor Recirculation Pumps, the failure of HPCI and RCIC to automatically start on low reactor water level, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. Applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed, including the manual start of HPCI and RCIC Systems.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches after manually starting HPCI and RCIC.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/20/00 **Test Number:** 5258

Test Title: Failure of the Automatic Reactor Trip System (MF-RP158003, MF-FW145009A, B, and C, Failure of RPS to Scram on a Loss of All Feedwater Pumps)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RP158003 (Failure to Scram) and MF-FW145009A, B, C (Reactor Feedwater Pumps A, B, and C Trip) were inserted to initiate the test.

Summary of Test:

The test verified the rapid reduction of reactor water level on the trip of all Reactor Feedwater Pumps, the failure of RPS to initiate a reactor scram on low reactor water level, the ability of the Alternate Rod Insertion (ARI) System to shut down the reactor, and the automatic start of HPCI and RCIC on low reactor water level. Following the reactor shutdown by the ARI System, the applicable operator actions of Procedure ON-100-101 were performed. After HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed.

Termination Criteria:

The reactor is shut down (all control rods have been inserted via ARI initiation) and reactor water level is being maintained above +13 inches with HPCI, RCIC, and the Condensate System as necessary.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/17/98 **Test Number:** 5259

Test Title: Reactor Pressure Control System Failure (MF-TC193013 and MF-TC193014, Loss of Turbine Primary and Backup Speed Signals)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-TC193013 (Loss of Turbine Primary Speed Signal) was inserted to initiate the test.

Summary of Test:

The test verified the swap from the Primary to the Backup Main Turbine Speed Control Circuit. After the control circuit swap, MF-TC193014 (Loss of Turbine Backup Speed Signal) was inserted. The test verified the total loss of Main Turbine Speed Control, the reactor pressure increase due to the trip of the Main Turbine, and the presence of appropriate annunciators. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with Feedwater System.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/17/98 **Test Number:** 5260

Test Title: Reactor Pressure Control System Failure Including Bypass Failure (MF-TC193012, Loss of 24 VDC Logic Bus Power)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-TC193012 (Loss of 24 VDC Logic Bus Power) was inserted to initiate the test.

Summary of Test:

The test verified the simulator's response to a Main Turbine trip without Bypass Valves available, the resultant reactor scram and Reactor Recirculation Pump trips, SRV actuations on high reactor pressure. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. After HPCI started, the applicable operator actions of Emergency Operating Procedure EO-100-102 (RPV Control Bases) were performed, including manually controlling reactor pressure with SRVs.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with the Reactor Feedwater System.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 08/18/98 **Test Number:** 5261

Test Title: Reactor Pressure Control System Failure (MF-TC193003, Turbine Pressure Regulator Oscillation)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-TC193003 (Turbine Pressure Regulator Oscillation) was inserted to initiate the test.

Summary of Test:

The test verified the reactor pressure and reactor power oscillations and the presence of appropriate annunciators. The manual transfer to the backup EHC pressure regulator without reduction of the pressure oscillations, the reduction of reactor power to 95 percent power, and the manual opening of one Bypass Valve using the LOAD LIMIT SET potentiometer per Off-Normal Procedure ON-193-001 (Turbine EHC System Malfunction) were performed. The test verified reactor pressure oscillations continued to diverge until a reactor scram occurred on pressure. Following the reactor scram, the applicable operator actions of Procedure ON-100-101 were performed. The Bypass Valve jack was used to manually open one BPV to reduce reactor pressure below 920 psig and stop the reactor pressure oscillations.

Termination Criteria:

Reactor water level is being maintained between +13 and +54 inches with the Reactor Feedwater System, and the reactor pressure oscillations have subsided.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 08/20/01 **Test Number:** 5301

Test Title: Transient Performance – Manual Scram

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. The RPS Manual Scram Channel A1 HS-C72A-1S03A and RPS Manual Scram Channel B1 HS-C72A-1S03B pushbuttons were depressed and the reactor mode switch was placed to the SHUTDOWN position to initiate the test.

Summary of Test:

The test verified the decrease of reactor power, the decrease of reactor pressure, the decrease of reactor water level due to void collapse, the runback of the Reactor Recirculation Pumps, the trip of the Main Turbine on Generator reverse power, reactor feedwater control setpoint setdown, and the presence of appropriate Control Room annunciators.

Termination Criteria:

At least three minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/20/01 **Test Number:** 5302

Test Title: Transient Performance – Simultaneous Trip of All Feedwater Pumps (MF-FW145009A, B, and C Reactor Feed Pump Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-FW145009A, B, and C (Reactor Feedwater Pump Trip) were inserted simultaneously to initiate the test.

Summary of Test:

The test verified the decrease of feedwater flow and pressure, the reactor scram on low reactor water level, the automatic start of HPCI and RCIC and the trip of the Recirc Pumps on low reactor water level, the MSIV isolation on low reactor pressure in RUN mode, the trip of HPCI and RCIC on high reactor water level, and the presence of appropriate Control Room annunciators.

Termination Criteria:

At least six minutes have elapsed.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/20/01 **Test Number:** 5303

Test Title: Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves (Override NSSSS)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Component Level Failures of B21-1K7A, B (N4S Relays) were simultaneously inserted to initiate the MSIV closure test.

Summary of Test:

The test verified the closure of the inboard and outboard MSIVs, the reactor scram on the MSIV closure, the decrease of reactor water level due to the void collapse, the trip of the Reactor Recirculation Pumps and automatic initiation of HPCI and RCIC on low reactor water level, the restoring of reactor water level by HPCI, RCIC, and the assistance of the Reactor Feedwater Pumps until driving steam was depleted, the control of reactor pressure by SRVs, and the presence of appropriate Control Room annunciators.

Termination Criteria:

The test was terminated after initial SRV lift occurred and a total of at least five minutes had elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 08/20/01 **Test Number:** 5304

Test Title: Transient Performance – Simultaneous Trip of All Recirculation Pumps (MF-RR164019A and B, Recirculation Pump Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-RR164019A, B (Recirculation Pumps A and B Trip) were simultaneously inserted to initiate the test.

Summary of Test:

The test verified the trip of the Reactor Recirculation Pumps, the decrease of core flow and jet pump flows, the rapid decrease of reactor power due to the core flow decrease, the decrease of Main Steam flow and Generator MWe, the reactor water level swell due to the core flow decrease, and the presence of appropriate Control Room annunciators.

Termination Criteria:

At least four minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/24/01 **Test Number:** 5305

Test Title: Transient Performance – Single Recirculation Pump Trip (RR164019A, Recirculation Pump "A" Trip)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR164019A (Recirculation Pump "A" Trip) was inserted to initiate the test.

Summary of Test:

The test verified the trip of the "A" Recirculation Pump, the core flow decrease, the "A" Recirc loop jet pump flow decrease, then increase as reverse flow was established, the reduction of reactor power and Generator MWe, the increase of the "B" Recirc loop jet pump flow, the reactor water level swell, the stabilizing of reactor power, and the presence of appropriate Control Room annunciators.

Termination Criteria:

At least six minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/24/01 **Test Number:** 5306

Test Title: Transient Performance – Main Turbine Trip (Maximum Power Level Which Does Not Result in an Immediate Reactor Scram – EG198001, Synch Output Breaker Trip)

Initial Conditions:

This test was performed from approximately 20 percent reactor power startup Xenon condition. Malfunction MF-EG198001 (Synch Output Breaker Trip) was inserted to trip the Main Turbine.

Summary of Test:

The test verified the appropriate Main Turbine Valve response, including the fast opening of the Bypass Valves, the absence of a reactor scram due to the Main Turbine trip, the decrease of feedwater inlet temperature and resultant reactor power increase, and the presence of the appropriate Control Room annunciators.

Termination Criteria:

Reactor power level has stabilized, and at least eight minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/22/01 **Test Number:** 5307

Test Title: Transient Performance – Maximum Power Ramp (Master Recirc Controller in Manual) Down to Approximately 75 Percent and Back Up to 100 Percent (MF-RR164005, RR Control Failure)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-RR164005 (Master Reactor Recirc Flow Controller XY-B31-1R620 Failure) at a 0 percent severity (minimum output) was inserted to ramp reactor power downward.

Summary of Test:

The first portion of the test verified the reduction of Recirc Pumps A and B speed, the decrease of total core flow and resultant reduction of reactor power to approximately 70 percent power, the EHC System response to control reactor pressure, the decrease of Main Generator output (MWe), and the ability of Feedwater Level Control System to control reactor water level.

After two minutes, Malfunction MF-RR164005 (Master Reactor Recirc Flow Controller XY-B31-1R260 Failure) was removed. This portion of the test verified the increase of core flow and resultant increase of reactor power including the temporary overshoot above 100 percent power, the EHC System response to control reactor pressure, the increase of Main Generator output (MWe), and the ability of Feedwater Level Control System to control reactor water level.

Termination Criteria:

Reactor water level has stabilized following the power increase, and at least four minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/24/01 **Test Number:** 5308

Test Title: Transient Performance – Maximum Size Reactor Coolant System Rupture Combined With a Loss of Offsite Power (MF-RR164011B, MF-DS003007 and MF-DS003008, LOCA With LOSP)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunctions MF-RR164011B (Recirc Loop B Suction Line Break at 100 percent severity (4.17 sq ft), DS003007 (Startup Transformer T-10 Lockout Actuation), and DS003008 (Startup Transformer T-20 Lockout Actuation) were simultaneously inserted to initiate the test.

Summary of Test:

The test verified the loss of offsite power, the reactor automatic scram, the rapid decrease of reactor water level, the automatic start and loading of the Emergency Diesel Generators, the rapid increase in drywell and suppression chamber pressures, the decrease of reactor pressure, the start and injection of the Residual Heat Removal and Core Spray Pumps, the increase in reactor water level due to low pressure ECCS and reference leg flashing, and the presence of appropriate Control Room annunciators.

Termination Criteria:

Reactor water level has been restored to above +13 inches by the low pressure Emergency Core Cooling Systems, and at least eight minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 08/24/01 **Test Number:** 5309

Test Title: Transient Performance – Maximum Size Unisolable Main Steam Line Rupture (MF-MS183007, A Main Steam Line Leak/Rupture Inside Containment)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Malfunction MF-MS183007 (Main Steam Line "A" Leak/Rupture Inside Containment) at 100 percent severity (Double-Ended Shear of 24 Inch Pipe) was inserted to initiate the test.

Summary of Test:

The test verified the closure of the MSIVs on high steam line flow, the reactor scram on MSIV closure, the initial swell followed by a rapid decrease of reactor water level, the trip of the Main Turbine and Reactor Feedwater Pumps on the initial high water level, the automatic start of the Emergency Diesel Generators, the increase of drywell and suppression chamber pressures and temperatures, the automatic start and injection of Residual Heat Removal and Core Spray Systems, the increase of reactor water level after reactor pressure decreased, and the presence of appropriate Control Room annunciators.

Termination Criteria:

Reactor water level has been restored to above +13 inches by the low pressure Emergency Core Cooling Systems and Condensate System, and at least ten minutes have elapsed.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/24/01 **Test Number:** 5310

Test Title: Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves Combined With Single Stuck Open Safety/Relief Valve (Inhibit Hi Pressure ECCS)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition. Component Level B21-1K7A, B (N4S Relays) and Malfunctions MF-MS183010B ("B" Safety Relief Valve Stuck Open) at 100 percent severity (full open) and MF-HP152002 (HPCI Auto Start Failure) were inserted to initiate the test.

Summary of Test:

The test verified the closure of the inboard and outboard MSIVs, the reactor scram on the MSIV closure, the trip of the Reactor Recirculation Pumps and the automatic start of RCIC on low reactor water level, the increase of reactor pressure to open SRVs, the sticking open of one SRV, the subsequent injection of water into the RPV by the Condensate Pumps via the Feedwater System, and the presence of appropriate Control Room annunciators.

Termination Criteria:

Reactor water level is above +13 inches, and the Feedwater System is injecting into the RPV.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 09/05/01 **Test Number:** 5501

Test Title: Steady-State Performance – 30 Percent Power

Initial Conditions:

This test was performed from a 30 percent power initial condition.

Summary of Test:

The test verified the accuracy's of critical and non-critical steady-state simulated parameters for which reference plant information was available.

Termination Criteria:

At least four minutes have elapsed and all of the parameters have been recorded.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/06/01 **Test Number:** 5502

Test Title: Steady-State Performance – 75 Percent Power

Initial Conditions:

This test was performed from a 75 percent power initial condition.

Summary of Test:

The test verified the accuracies of critical and non-critical steady-state simulated parameters for which reference plant information was available.

Termination Criteria:

At least four minutes have elapsed and all of the parameters have been recorded.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/02/01 **Test Number:** 5503

Test Title: Steady-State Performance – 100 Percent Power and One Hour Stability Test

Initial Conditions:

This test was performed from a 100 percent power initial condition.

Summary of Test:

The test verified the accuracies and stability of critical and non-critical steady-state simulated parameters for which reference plant information was available.

Termination Criteria:

At least 60 minutes have elapsed since the initial conditions were established, and all the parameters have been recorded.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/06/01 **Test Number:** 6501

Test Title: Unit Two Steady-State Performance – 30 Percent Power

Initial Conditions:

This test was performed from a 30 percent power initial condition.

Summary of Test:

The test verified the accuracies of critical and non-critical steady-state simulated parameters for which reference plant information was available.

Termination Criteria:

At least four minutes have elapsed and all of the parameters have been recorded.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/13/01 **Test Number:** 6502

Test Title: Unit Two Steady-State Performance – 70 Percent Power

Initial Conditions:

This test was performed from a 70 percent power initial condition.

Summary of Test:

The test verified the accuracies of critical and non-critical steady-state simulated parameters for which reference plant information was available.

Termination Criteria:

At least four minutes have elapsed and all of the parameters have been recorded.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/13/01 **Test Number:** 6503

Test Title: Unit Two Steady-State Performance – 100 Percent Power

Initial Conditions:

This test was performed from a 100 percent power initial condition.

Summary of Test:

The test verified the accuracies of critical and non-critical steady-state simulated parameters for which reference plant information was available.

Termination Criteria:

At least 60 minutes have elapsed and all of the parameters have been recorded.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 07/28/98 **Test Number:** 5601

Test Title: Simulator Operating Limits (Drywell Floor dp Down and High Reactor Pressure)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition.

Summary of Test:

The first portion of the test was initiated by increasing the drywell mass. When the simulator was placed into RUN, the test verified the sudden pressure increase in the drywell, the Simulator Operating Limiter light illuminating when the Drywell floor downward differential pressure reached 28 psid, and the Simulator Operating Limit light remaining lit when Drywell floor downward differential pressure decreased below 28 psig.

The second portion of the test established a failure of the following: RPS to scram the reactor, Recirc Pump RPT breakers to open, Main Turbine Bypass Valves to open, and all Safety Relief Valves to open. While manually reducing the EHC Load Set setpoint, the test verified the Simulator Operating Limit light was illuminated when reactor vessel pressure reached 1,325 psig. While manually increasing the EHC Load Set setpoint, the test verified the Simulator Operating Limit light remaining lit when reactor pressure decreased below 1,325 psig.

Termination Criteria:

Ensure the Simulator Operating Limit light remains illuminated when the "Drywell Floor Downward Differential Pressure" and "Vessel Pressure High" operating limits were below their setpoints.

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 07/13/99 **Test Number:** 5602

Test Title: Simulator Operating Limits (High Containment Pressure and Temperature, External Pressure, and Differential Pressure (Up))

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition.

Summary of Test:

The first portion of the test inserted various malfunctions, opened Safety Relief Valves, and verified the Simulator Operating Limit light was illuminated when (1) Suppression Pool liquid temperature was above boiling and (2) Drywell floor upward differential pressure was above 5.5 psig. The malfunctions were removed and the test verified the Simulator Operating Limit light remaining lit when the operating limits were below their setpoints.

The second portion of the test required setting the parameter to its operating limit to verify the Simulator Operating Limit light and by means of inserted various malfunctions to verify the Operating Limit Summary displayed the proper status of the Operating Limit tested. The test verified the Simulator Operating Limit light was illuminated when (3) Suppression Chamber air temperature reached 220 °F, (4) Drywell internal temperature reached 340 °F, (5) Containment pressure was above 135 psig, and (6) Drywell external pressure reached 5 psid.

Termination Criteria:

Ensure the Simulator Operating Limit light remains illuminated when the operating limits were below their setpoints.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt
MANAGER – NUCLEAR TRAINING DATE: 09/14/01

Date Tested: 09/15/00 **Test Number:** 5603

Test Title: Simulator Operating Limits (Fuel Clad Temperature)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition.

Summary of Test:

Malfunction MF-RP158003 (Failure to Scram) and other malfunctions were inserted generating an ATWS condition without injection of water into the RPV. After reactor water level was reduced to below Top of Active Fuel (TAF), the ATWS malfunctions were removed, the reactor automatically scrammed, and all of the SRVs were manually opened. The test verified the Simulator Operating Limit light is illuminated when the maximum fuel clad temperature reaches 2500 °F. All malfunctions were then removed to allow injection of water into the RPV, and the Simulator Operating Limit light was verified to remain illuminated.

Termination Criteria:

After three minutes of injection of water into the RPV, ensure the Simulator Operating Limit light remains illuminated.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 08/04/01 **Test Number:** 5604

Test Title: Simulator Operating Limits (250 V, 125 V, and 24 VDC Buses)

Initial Conditions:

This test was performed from a 100 percent reactor power equilibrium Xenon condition.

Summary of Test:

Various remote functions were utilized to open feeder breakers to the battery chargers for each 250 V, 125 V, and 24 VDC battery. To reduce the time of battery voltage decay, the battery capacities were set to below 20 amperes one battery at a time. The test verified the battery voltage decayed to the point the bus de-energized and the Simulator Operating Limit light illuminated. The battery charger feeder breakers were closed one bus at a time. The test verified the battery voltage increased, the buses re-energized, and the Simulator Operating Limit light remained illuminated.

Termination Criteria:

After the battery voltage for each of the 250 V, 125 V, and 24 VDC batteries was restored to above the operating limit, the Simulator Operating Limit light remains illuminated RED.

TEST ABSTRACTS **SUSQUEHANNA STEAM ELECTRIC STATION**

PREPARED BY: F. Tarselli DATE: 09/14/01

APPROVED BY: W. W. Hunt DATE: 09/14/01
MANAGER – NUCLEAR TRAINING

Date Tested: 09/11/01 **Test Number:** 5610

Test Title: Simulator Computer Real Time Test

Initial Conditions:

This test was performed from 100 percent power equilibrium Xenon condition. Malfunction MF-MS183007 (A Main Steam Line Leak/Rupture Inside Containment) at 100 percent severity (Double-Ended Shear of 24 Inch Pipe) was inserted as the worst case transient to initiate the test.

Summary of Test:

The test was terminated after Real Time Status (RTS) and the System Resource Manage/Accounting Task (MEMTIM) to determine the simulator timing. The test verified the simulator computers do not exceed 100 percent of the time allotted for any given frame.

Termination Criteria:

The test was terminated after Real Time Status Program (RTS) and the System Resource Manager/Accounting Task (MEMTIM) data was captured.

Susquehanna Steam Electric Station, Unit One

Docket # 50 - 387

PPL Susquehanna, LLC

September 14, 2001

PERFORMANCE TEST SCHEDULE

Performance Test Schedule

The Simulator Certification performance testing schedule lists the test number, the applicable ANSI section(s) covered, and the title or description of the performance test. The schedule is arranged as follows:

- Simulator tests to be conducted annually.
- Additional simulator tests to be conducted the first year.
- Additional simulator tests to be conducted the second year.
- Additional simulator tests to be conducted the third year.
- Additional simulator tests to be conducted the fourth year.

Simulator Tests to be Conducted Annually:

5610	Appendix A3.1 Simulator Computer Real Time Test
5501	Appendix B1.1 Steady-State Performance – 30 Percent Power
5502	Appendix B1.1 Steady-State Performance – 75 Percent Power
5503	Appendix B1.1 Steady-State Performance – 100 Percent Power and One Hour Stability
6501	Appendix B1.1 Unit Two Steady-State Performance – 30 Percent Power
6502	Appendix B1.1 Unit Two Steady-State Performance – 70 Percent Power
6503	Appendix B1.1 Unit Two Steady-State Performance – 100 Percent Power
5301	Appendix B1.2 (1) Transient Performance – Manual Scram
5302	Appendix B1.2 (2) Transient Performance – Simultaneous Trip of All Feedwater Pumps
5303	Appendix B1.2 (3) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves
5304	Appendix B1.2 (4) Transient Performance – Simultaneous Trip of All Recirculation Pumps
5305	Appendix B1.2 (5) Transient Performance – Single Recirculation Pump Trip
5306	Appendix B1.2 (6) Transient Performance – Main Turbine Trip (Maximum Power Level Which Does Not Result in an Immediate Reactor Scram)
5307	Appendix B1.2 (7) Transient Performance – Maximum Power Ramp (Master Recirc Controller in Manual) Down to Approximately 75 Percent and Back Up to 100 Percent
5308	Appendix B1.2 (8) Transient Performance – Maximum Size Reactor Coolant System Rupture Combined With a Loss of Offsite Power
5309	Appendix B1.2 (9) Transient Performance – Maximum Size Unisolable Main Steam Line Rupture
5310	Appendix B1.2 (10) Transient Performance – Simultaneous Closure of All Main Steam Isolation Valves Combined With Single Stuck Open Safety/Relief Valve (Inhibit Activation of High Pressure ECCS)

(Continued)

Additional Simulator Tests to be Conducted the First Year:

5601	Section 4.3 Simulator Operating Limits (Drywell Floor dp Down and High Reactor Pressure)
5101	3.1.1 (1, 5) Plant Startup – Cold to Hot Standby (Including Operations at Hot Standby)
5102	3.1.1 (2, 3) Nuclear Startup from Hot Standby to Rated Power (Including Turbine Startup and Generator Synchronization)
5202	3.1.2 (1) Small Reactor Coolant Breaks Outside Primary Containment ("D" Main Steam Line Leak Outside Containment TB Pipe Tunnel at Two Percent Severity)
5203	3.1.2 (1) Small Reactor Coolant Breaks Inside Primary Containment (Reactor Vessel Bottom Head Drain Line Leakage)
5205	3.1.2 (2) Loss of Instrument Air (Instrument Air Leakage – Turbine Building Elevation 646')
5206	3.1.2 (2) Loss of Instrument Air (Instrument Air Common Header Rupture)
5207	3.1.2 (2) Loss of Instrument Air (Instrument Air Leakage – Turbine Building Elevation 699')
5215	3.1.2 (3) Loss or Degraded Electrical Power to the Station (Startup Bus 10 Lockout)
5216	3.1.2 (3) Loss or Degraded Electrical Power to the Station (Startup Buses 10 and 20 Lockout and Generator Synch Output Breaker 1R101 Spurious Trip)
5217	3.1.2 (3) Loss or Degraded Electrical Power to the Station (Auxiliary Bus 11A – Fails to Fast Transfer)
5218	3.1.2 (3) Loss or Degraded Electrical Power to the Station (4.16 kV ESS Bus 1B Differential Current Lockout Trip and Diesel Generator "B" Fails to Start)
5219	3.1.2 (3) Loss or Degraded Electrical Power to the Station (13.8 kV Bus 11B Lockout)
5224	3.1.2 (7) Loss of Shutdown Cooling (RHRSW Loop 1A Leak/Rupture)
5229	3.1.2 (9) Normal Feedwater System Failure (Reactor Feed Pump C Trip)
5230	3.1.2 (9) Normal Feedwater System Failure (Loss of Extraction Steam to Feedwater Heater 4B)
5231	3.1.2 (10) Loss of All Feedwater (Normal and Emergency) (RCIC Turbine Trip, HPCI Auto Start Failure, Reactor Feed Pumps A, B, and C Trip)

5232	3.1.2 (11) Loss of Protective System Channel (Failure to Scram – Low Power Natural Circulation MSIVs Closed ATWS)
5233	3.1.2 (11) Loss of Protective System Channel (Failure to Scram – High Power MSIVs Open ATWS)
5239	3.1.2 (15) Turbine Trip (Main Turbine Trip)
5240	3.1.2 (16) Generator Trip (Negative Phase Sequence Trip – Generator Load Reject)
5245	3.1.2 (20) Main Feed Line Break Outside Containment (Feedwater Discharge Line "A" Leak Outside Containment)
5259	3.1.2 (25) Reactor Pressure Control System Failure Including Turbine Bypass Failure (Loss of Turbine Primary and Backup Speed Signals)
5260	3.1.2 (25) Reactor Pressure Control System Failure Including Turbine Bypass Failure (Loss of 24 VDC Logic Bus Power)
5261	3.1.2 (25) Reactor Pressure Control System Failure (Turbine Pressure Regulator Oscillation)

Additional Simulator Tests to be Conducted the Second Year:

5602	Section 4.3 Simulator Operating Limits (High Containment Pressure and Temperature, External Pressure, and Differential Pressure (Up))
5106	3.1.1 (8, 6) Plant Shutdown from Rated Power to Hot Standby (Including Load Changes) Followed by a Plant Cooldown to Cold Shutdown Conditions
5108	3.1.1 (9) Shutdown Margin Determination
5109	3.1.1 (9) In-Sequence Critical and Shutdown Margin Demonstration
5201	3.1.2 (1) Large Reactor Coolant Breaks Outside Primary Containment ("D" Main Steam Line Leak Outside Containment TB Pipe Tunnel at 100 Percent Severity)
5204	3.1.2 (1) Failure of Safety Relief Valves (Safety Relief Valve "M" Fails Open)
5208	3.1.2 (3) Loss or Degraded Electrical Power to the Station (Sustained Station Blackout With Generator Output Breaker Trip)
5220	3.1.2 (5) Loss of Condenser Vacuum Including Loss of Condenser Level Control (Main Condenser Tube Leak at One Percent Severity)
5221	3.1.2 (5) Loss of Condenser Vacuum Including Loss of Condenser Level Control (Main Condenser Water Level Control Failure – High)
5222	3.1.2 (5) Loss of Condenser Vacuum Including Loss of Condenser Level Control (Circ Water Pumps A, B, C, and D Trip)
5223	3.1.2 (6) Loss of Service Water (Service Water Pumps A, B, and C Trip)
5234	3.1.2 (12) Control Rod Failure Including Drifting Rods and Misaligned Rods (Control Rod 22-27 Drift In)
5235	3.1.2 (12) Control Rod Failure Including Drifting Rods and Misaligned Rods (Control Rod 26-35 Drift Out)
5236	3.1.2 (12) Control Rod Failure Including Stuck Rods, Uncoupled Rods, and Dropped Rod (Control Rod 26-35 Stuck and Uncoupled)
5237	3.1.2 (13) Inability to Drive Control Rods (RMCS Sequence Timer Malfunction)
5238	3.1.2 (14) Fuel Cladding Failure Resulting in High Activity in Reactor Coolant (Fuel Cladding Failure)

5241 3.1.2 (17) Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal (Actuation of the Recirc "B" 30 Percent Limiter)

5242 3.1.2 (17) Failure in Automatic Control System(s) That Affect Reactivity and Core Heat Removal (Inadvertent (Spurious) HPCI Initiation)

Additional Simulator Tests to be Conducted the Third Year:

5603 Section 4.3 Simulator Operating Limits (Fuel Clad Temperature)

5103 3.1.1 (4) Reactor Trip Followed by Recovery to Rated Power

5107 3.1.1 (9) Plant Heat Balance

5209 3.1.2 (3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D612)

5210 3.1.2 (3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D622)

5211 3.1.2 (3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D632)

5212 3.1.2 (3) Loss or Degraded Electrical Power to the Station (Loss of 125 VDC Bus 1D642)

5213 3.1.2 (3) Loss or Degraded Electrical Power to the Station (Loss of AC Instrument Bus 1Y218/1Y219)

5214 3.1.2 (3) Loss or Degraded Electrical Power to the Station (Loss of AC Instrument Bus 1Y226)

5250 3.1.2 (22a) Process Instrumentation System Failure (Recirc Flow Unit A Failure)

5251 3.1.2 (22b) Process Alarm System Failure (Loss of Annunciator 1C651 - AR101)

5252 3.1.2 (22b) Process Alarm System Failure (Loss of Annunciator 1C651 - AR110)

5253 3.1.2 (22b) Process Alarm System Failure (Loss of Annunciator OC653 - AR015)

5254 3.1.2 (22c) Process Control System Failure (Recirc Pump A M/A Station Auto Output Failure – High)

5255	3.1.2 (22c) Process Control System Failure (Feedwater Master Level Controller Failure – High)
5256	3.1.2 (23) Passive Malfunction in Emergency Feedwater (Loss of Core Spray Division 2 Initiation Logic Power and Recirc Loop B Discharge Line Break)
5257	3.1.2 (23) Passive Malfunction in Emergency Feedwater (RCIC and HPCI Automatic Start Failure When All Feedwater Pumps Trip)
5258	3.1.2 (24) Failure of the Automatic Reactor Trip System (Failure to Scram on a Loss of All Feedwater Pumps)

Additional Simulator Tests to be Conducted the Fourth Year:

5604	Section 4.3 Simulator Operating Limits (250 V, 125 V, and 24 VDC Buses)
5104	3.1.1 (7a, c) Startup and Power Operations With Less Than Full Reactor Coolant Flow
5105	3.1.1 (7b, c) Shutdown and Power Operations With Less Than Full Reactor Coolant Flow
5110	3.1.1 (10) Operator Conducted Surveillance Testing on Safety-Related Equipment or Systems
5225	3.1.2 (8) Loss of Component Cooling System or Cooling to Individual Components (CRD Pump A Bearing and Lube Oil Cooler Flow Blockage)
5226	3.1.2 (8) Loss of Component Cooling System or Cooling to Individual Components (Reactor Recirculation "A" M/G Set Hydraulic Fluid Cooler Flow Blockage)
5227	3.1.2 (8) Loss of Component Cooling System or Cooling to Individual Components (Trip of All Drywell Cooling Fans)
5228	3.1.2 (8) Loss of Component Cooling System or Cooling to Individual Components (Trip of All Drywell Cooling Fans)
5243	3.1.2 (19) Reactor Trip (Control Rods in Scram Group No. 1 Receive a Scram Signal – 45 Control Rods Scrammed)
5244	3.1.2 (20) Main Feed Line Break Inside Containment (Feedwater Discharge Line "B" Leak/Rupture Inside Containment)
5246	3.1.2 (21) Nuclear Instrumentation Failure (SRM Channel A Failure)
5247	3.1.2 (21) Nuclear Instrumentation Failure (IRM Channel B Failure)
5248	3.1.2 (21) Nuclear Instrumentation Failure (APRM Channel C Failure)
5249	53.1.2 (21) Nuclear Instrumentation Failure (Rod Block Monitor Channel A Failure)

NRC FORM 474 U.S. NUCLEAR REGULATORY COMMISSION (8-1998) **SIMULATION FACILITY CERTIFICATION**	**APPROVED BY OMB: NO. 3150-0138EXPIRES: 08/31/2001** Estimated burden per response to comply with this mandatory information collection request: 120 hours. This information is used to certify a simulation facility. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0138), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

INSTRUCTIONS: This form is to be filed for initial certification, recertification (if required), and for any change to a simulation facility performance testing plan made after initial submittal of such a plan. Provide the following information and check the appropriate box to indicate reason for submittal.

Susquehanna Steam Electric Station – Unit 2	DOCKET NUMBER 50-388
LICENSEE PPL Susquehanna, LLC	DATE 09/14/01

This is to certify that:
1. The above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR 55.45.
2. Documentation is available for NRC review in accordance with 10 CFR 55.45(b).
3. This simulation facility meets the guidance contained in ANSI/ANS 3.5-1985 or ANSI/ANS 3.5-1993, as endorsed by NRC Regulatory Guide 1.149.

If there are any **EXCEPTIONS** to the certification of this item, **CHECK HERE** [X] and describe fully on additional pages as necessary.

NAME *(or other identification)* AND LOCATION OF SIMULATION FACILITY.

SSES Simulator
Route 11, Approximately Five Miles North of Berwick, PA

X SIMULATION FACILITY PERFORMANCE TEST ABSTRACTS ATTACHED. *(For performance tests conducted in the period ending with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING COMPLETED. *(Attach additional pages as necessary and identify the item description being continued.)*

See Page 6 for Performance Test Corrections made to performance Tests Conducted Since Original Certification.

X SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. *(For the conduct of approximately 25 percent of performance tests per year for the four-year period commencing with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING TO BE CONDUCTED. *(Attach additional pages as necessary and identify the item description being continued.)*

X PERFORMANCE TESTING PLAN CHANG. *(For any modification to a performance testing plan submitted on a previous certification.)*

DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE *(Attach additional pages as necessary and identify the item description being continued.)*

[ ] RECERTIFICATION *(Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR 55.45(b)(5)(v). (Attach additional pages as necessary and identify the item description being continued.)*

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE AUTHORIZED REPRESENTATIVE	TITLE	DATE
	Senior Vice President and Chief Nuclear Officer	09/14/01

In accordance with 10 CFR 55.5, Communications, this form shall be submitted to the NRC as follows:

BY MAIL ADDRESSED TO:	**DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION U.S. NUCLEAR REGULATORY COMMISSION WASHINGTON, DC 20555-0001**	BY DELIVERY IN PERSON TO THE NRC OFFICE AT:	**ONE WHITE FLINT NORTH 11555 ROCKVILLE PIKE ROCKVILLE, MD**

NRC FORM 474 (8-1998) PRINTED ON RECYCLED PAPER

Susquehanna Steam Electric Station, Unit Two

Docket #50 - 388

PPL Susquehanna, LLC

September 14, 2001

EXCEPTIONS

Exceptions

Certification Exception 005 is as listed in the original certification document dated September 27, 1993.

Certification Exception 006 is listed in the original certification document dated September 27, 1993