NRC FORM 474 (1-90) 10 CFR 55.45(b), 55.4 and 55.5

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0138 EXPIRES 9-30-92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 120 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0138), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

SIMULATION FACILITY CERTIFICATION

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.

FACILITY: Perry Nuclear Power Plant, Unit No. 1

DOCKET NUMBER: 50-440

LICENSEE: The Cleveland Electric Illuminating Co. and Centerior Service Co., et al.

DATE: 6/26/91

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, as endorsed by NRC Regulatory Guide 1.149. If there are any exceptions to the certification of this item, check here [✓] and describe fully on additional pages as necessary.

Please refer to Tab B "Exceptions to ANS 3.5"

NAME (for other identification) AND LOCATION OF SIMULATION FACILITY

Perry Nuclear Power Plant Unit 1 Simulator
10 Center Road
Perry, OH 44081

(Please refer to Tab A "Certification Overview" for additional simulator information.)

[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 page(s) as necessary, and identify the item description being continued)*

Please refer to Tab C "Simulator Tests"

[X] SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. *(For the conduct of approximately 25% 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 page(s) as necessary, and identify the item description being continued)*

Please refer to Tab D "Four Year Test Schedule"

[ ] 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 page(s) as necessary, and identify the item description being continued)*

N/A

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

N/A

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]	Vice President - Nuclear	6/28/91

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

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: One White Flint North, 11555 Rockville Pike, Rockville, MD

TAB A CERTIFICATION OVERVIEW

Part 1. Simulator Information

Part 2. Simulator Design Data

Part 3. Simulator Discrepancy Resolution and Upgrading

TAB B EXCEPTIONS TO ANS 3.5

Part 1. Permanent Exceptions

Part 2. Temporary Exceptions (Discrepancies)

Part 3. Schedule for Correcting Discrepancies

TAB C SIMULATOR TESTS

Part 1. Certification Test to ANSI/ANS 3.5 Section Cross Reference

Part 2. Computer Real Time Test Abstract

Part 3. Steady State and Normal Operations Test Abstracts

A. Core Performance Tests
B. Continuous Plant Operation Tests
C. Steady State Performance Tests

Part 4. Transient Performance Test Abstracts

Part 5. Malfunction Test Abstracts

A. System Level Failures
B. Malfunction Scenarios

TAB D FOUR YEAR TEST SCHEDULE

On the Cover:

Plant and simulator milestones and history	*Completion of hardware installation at S3 Technologies in Columbia, MD*
Factory testing at S3 Technologies	*Training and Education Center at Perry.*

PERRY POWER PLANT
SIMULATION FACILITY CERTIFICATION
JUNE 1991

TAB A CERTIFICATION OVERVIEW

Part 1. Simulator Information

A. General Information

B. Control Room Physical Fidelity

C. Instructor Interface

D. Operating Procedures

Part 2. Simulator Design Data

Part 3. Simulator Discrepancy Resolution and Upgrading

A. GENERAL INFORMATION

1. Owner/Operator/Manufacturer

Owners: Cleveland Electric Illuminating Company
10 Center Road
Perry, OH 44081

Centerior Service Company
6200 Oaktree Blvd.
Independence, OH 44131

Operator: Cleveland Electric Illuminating Company
Perry Nuclear Power Plant
Nuclear Training Section

Manufacturer: S3 Technologies Co.
8930 Stanford Blvd.
Columbia, MD 21045-4752

Note: Former company names which may appear in this document include Singer, Link-Miles Simulation, and Singer Link-Miles.

2. Reference Plant/Type/Rating

Reference Plant: Perry Nuclear Power Plant, Unit No. 1

Type: Boiling Water Reactor, GE

Rating: 3579 MWth

3. Date Ready For Training

After completion of Factory Acceptance Testing, delivery, installation, and site acceptance testing which is scheduled for completion before August 31, 1991.

4. Type of Report

Initial Certification

5. Acronyms Used Throughout the Report

ATP - Acceptance Test Procedure
CMS - Configuration Management System (computer system)
CUN - Configuration Update Notice (plant drawing change)
DCP - Design Change Package (plant change document package)
DCN - Drawing Change Notice (plant drawing change)
IC - Initial Conditions
PE - Permanent Exception to ANSI/ANS 3.5
PEI - Plant Emergency Instructions
SCR - Setpoint Change Request
SDP - Simulator Discrepancy Package (CEI document)
SDR - Simulator Discrepancy Report (S3 Technologies document)
TE - Temporary Exception to ANSI/ANS 3.5
TIE - Training Impact Evaluation
TGIS - Third Generation Instructor's Station

5. Acronyms (continued)

TMA - Training Manual Administrative Instruction
TSR - Training Significance Review

B. CONTROL ROOM PHYSICAL FIDELITY

1. The physical scope of the plant-referenced simulator was established in accordance with section 3.2 of ANSI/ANS 3.5-1985 with the active participation of two SRO's, one from plant operations and one from operator training. Tab A, Part 1, Attachments 1 and 2 illustrate the physical arrangement of both the Simulator Room and Control Room panels, respectively. Those Control Room panels included in the Simulator design were the only panels determined to be required for conducting normal plant evolutions and for responding to the malfunctions listed in ANSI/ANS 3.5 section 3.1.2. Panels not included in the Simulator Room were evaluated as having no or minimal training impact. These as well as other simulator design limitations which were established as part of the Simulator scoping process are listed in Tab B, Part 1 - "Permanent Exceptions to ANS 3.5."

2. Physical fidelity of the controls on the simulated panels, Ref. ANSI/ANS 3.5 section 3.2.2, has been verified against design-freeze photographs (dated July 1, 1989) of Perry Unit 1 panels in the scope of the plant-referenced simulator.

3. Design changes which physically affect Perry Unit 1 have been evaluated since the July 1, 1989 design-freeze date through the Training Significance Review (TSR) process (Ref. Tab A, Part 3, Section B). The plant changes which affect panels that are simulated have been handled as follows:

   a. Change has been or will be incorporated in the plant-referenced simulator prior to the time it is declared ready for training. Photographs of Perry Unit 1 are used in verifying the physical fidelity of the plant-referenced simulator with respect to the plant change.

   b. Change remains as outstanding work to be completed before June 28, 1992 and is documented under Tab B, Part 2, Section B - "Temporary Exceptions to ANS 3.5" (post design freeze discrepancies).

4. The ongoing program to ensure physical fidelity of the plant-referenced simulator consists of the TSR process (Tab A, Part 3, Section B) and PNPP Simulator Element Desk Guide 015 "Environmental Comparison of Simulator to Unit 1" which provides for periodic direct examination of physical fidelity of the simulator.

C. INSTRUCTOR INTERFACE

1. Description

The Perry Unit 1 Simulator Instructor Interface consists of an S3 Technologies Third Generation Instructor's Station (TGIS) version

C. **INSTRUCTOR INTERFACE (continued)**

4.1, which incorporates the Advanced Man-Machine Interface (AMMI) software design. The TGIS consists of 3 fixed Workstations and 1 roll-around instructors station, one color copier, one 8 channel strip chart recorder, one eight-pen color plotter, a Laser Printer, two types of remote transmitters (full feature and limited feature), and various peripherals. The Instructors Station computer system is independent of the main simulation computer complex. An interface is provided to permit direct memory access between the two computer systems.

2. Simulator Training Capabilities

2.1 Initial Conditions

The Simulator possesses the capability to store 230 Initial Conditions. 200 of the IC's are reserved for training use and 30 are designated as "discrepancy" snapshots which an instructor can use to store up to 15 minutes of operating data when a problem is observed. 50 IC's are designated as double password protected permanent IC's. 22 IC's are delivered from the factory for immediate use by the Perry Training Staff. These 22 IC's include a variety of operating conditions, fission product poison concentrations, and various times in core life.

2.2 Malfunctions

Malfunctions can be conveniently inserted by the instructor from either the fixed workstations or the hand-held remote devices. A series of malfunctions can be inserted (up to 300 discrete failures) simultaneously or sequentially using time delays and triggers. The capability for adding additional malfunctions is a basic design feature of the simulator. Attachment 3 to TAB A, Part 1 lists the generic malfunctions included in the 4 year performance test plan. TAB D is the schedule of testing for the next 4 years. The individual test abstracts (TAB C) list the malfunctions included in each test.

2.3 Other Control Features

The Perry Simulator design includes capabilities to freeze simulation, run simulation in variable slow time, and backtrack. Fast time is a selectable feature for various parameters and evolutions, such as xenon concentration, turbine coastdown, and drawing a vacuum. The TGIS includes many other useful features not required by ANSI/ANS 3.5. A series of tests were run at the factory to ensure all features work as intended. Documentation of testing and all design specifications are available for review.

C. **INSTRUCTOR INTERFACE (continued)**

2.4 Simulation of Items Outside the Control Room

The simulation complex was designed to provide as much realism as possible in the way the trainee interfaces to the systems and components located outside the Control Room. The main instructor control station includes a communication system which allows the instructor to communicate with the trainees exactly as they would communicate with local operators, including phones, radios and the GAI-Tronics internal communication system. The remote functions available allow for complete simulation of local operator actions required to perform the evolutions listed in Section 3.1 of ANSI/ANS 3.5. Remote functions can be controlled from the fixed work stations or keyed from the hand-held transmitter devices.

D. **OPERATING PROCEDURES**

Reference plant controlled procedures are used in the simulator. There are no differences.

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JUN 12 1991
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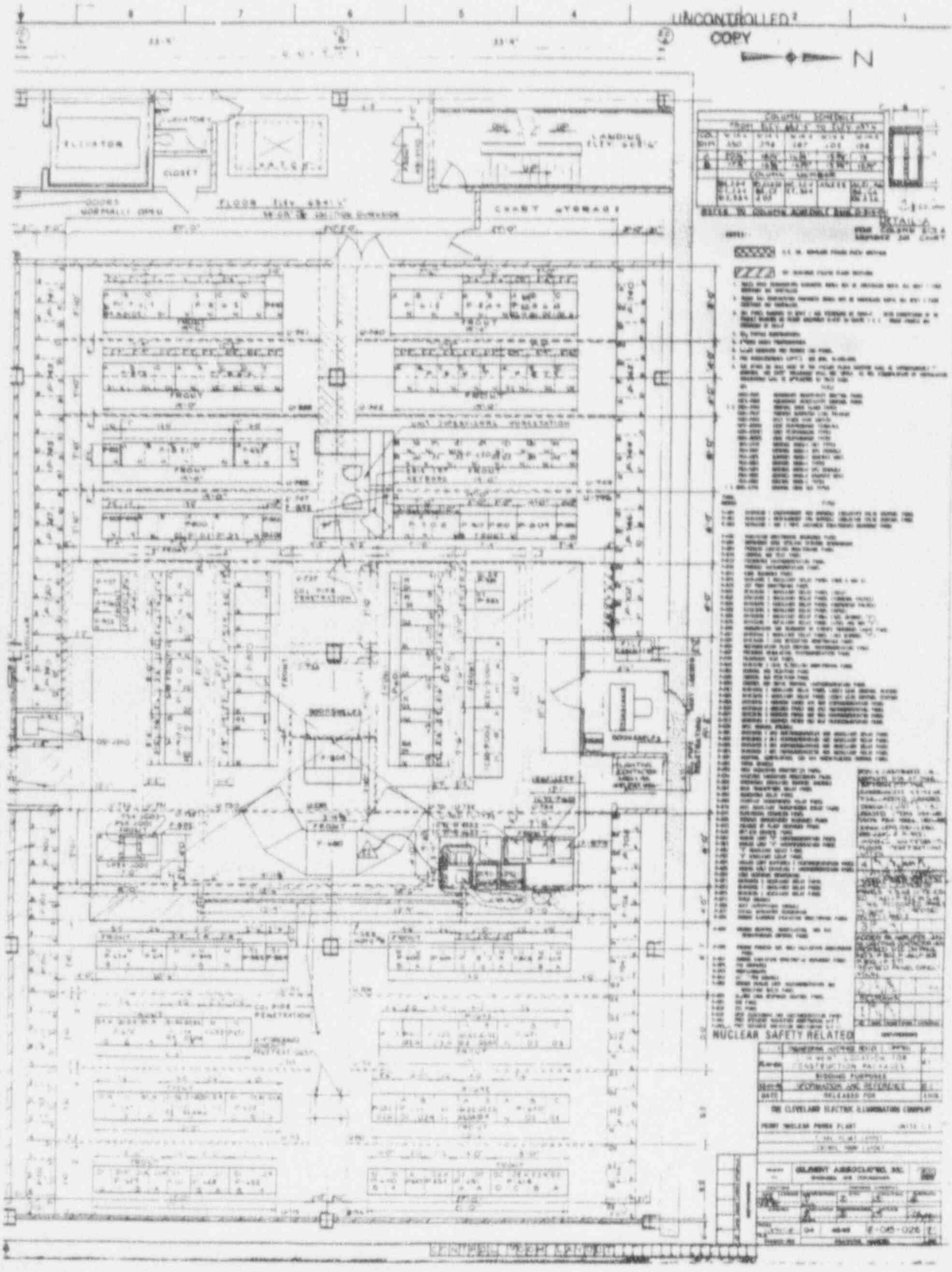
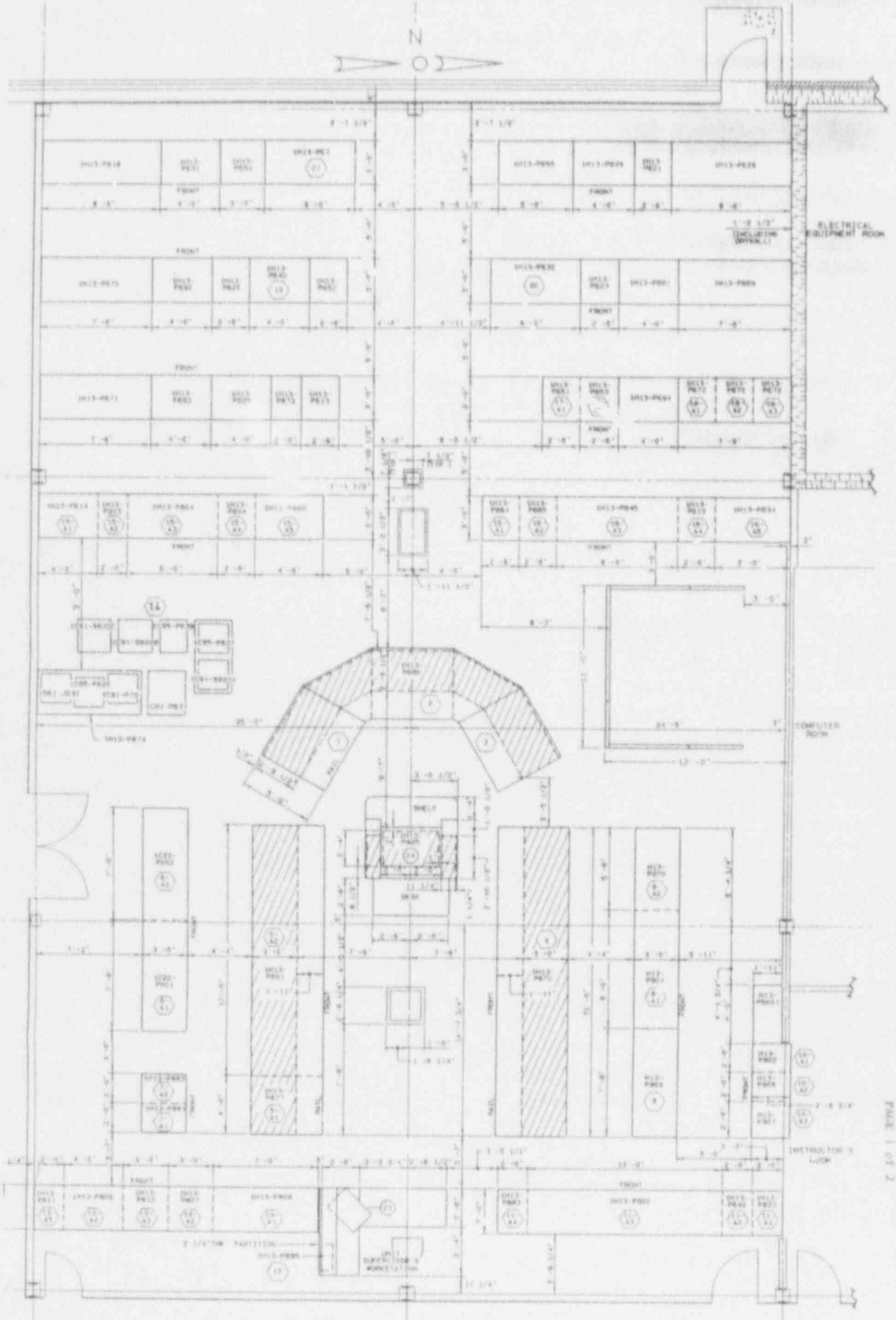


CERTIFICATION OF PERRY SIMULATION FACILITY
Certification Overview - Simulator Information

DOCKET NO. 50-440
TAB A, Part 1 - Attachment 2
PAGE 1 of 1



SIMULATOR ROOM 123 FLOOR PLAN

CERTIFICATION OF PERRY SIMULATION FACILITY
Certification Overview - Simulator Information

DOCKET NO. 50-440
TAB A, Part 1 - Attachment 1
PAGE 1 of 2

SINGER FURNISHED EQUIPMENT SCHEDULE

IT NO.	CONTROL RM MPL NO.	DESCRIPTION
5-A1	1C22-P001	REDUNDANT REACTIVITY CONTROL PANEL
6-A2	1C22-P002	REDUNDANT REACTIVITY CONTROL PANEL
14-A3	1C91-P637	PROCESS COMPUTER LINE PRINTER
15-A5	1H13-P600	RADIATION MONITORING RECORDER PANEL
5-A2	1H13-P601	EMERGENCY CORE COOLING SYSTEMS BENCHBOARD
15-A4	1H13-P604	PROCESS RADIATION MONITORING PANEL
15-A1	1H13-P614	NSSS RECORDER PANEL
16-A4	1H13-P619	JET PUMP MONITORING PANEL
20	1H13-P632	LEAK DETECTION PANEL-DIVISION 1
16-A5	1H13-P634	RECIRCULATION FLOW CONTROL INSTRUMENTATION PANEL
19	1H13-P642	LEAK DETECTION PANEL-DIVISION 2
17-A1	1H13-P651	CONTROL ROD POSITION PANEL
17-A2	1H13-P653	CONTROL ROD DRIVE CONTROL INSTRUMENTATION PANEL
18-A1	1H13-P672	NEUTRON/PROCESS RAD MON NON DIV
18-	1H13-P672	PROCESS RAD MON SRM & IRM CH. D/M
18-	1H13-P672	POWER RANGE NEUTRON MON CH. D/M
1	1H13-P680	UNIT CONTROL CONSOLE
2	1H13-P680	UNIT CONTROL CONSOLE
3	1H13-P680	UNIT CONTROL CONSOLE
11-A3	1H13-P800	HEATING, VENTILATION & AIR CONDITIONING CONTROL PANEL
15-A2	1H13-P803	AREA RADIATION MONITORING PANEL
15-A3	1H13-P804	AIRBORNE RADIATION MONITORING PANEL
13-A2	1H13-P807	MAIN TRANSFORMER RELAY PANEL
13-A1	1H13-P808	GENERATOR RELAY PANEL
13-A4	1H13-P809	START-UP TRANSFORMER RELAY PANEL
13-A3	1H13-P810	UNIT AUXILIARY TRANSFORMER RELAY PANEL
13-A5	1H13-P811	ELECTRICAL RECORDER PANEL
11-A1	1H13-P823	TURBINE SUPERVISORY RECORDER PANEL
11-A2	1H13-P842	BALANCE OF PLANT RECORDER PANEL
16-A3	1H13-P845	OFF-GAS CONTROL PANEL
4	1H13-P870	LONG RESPONSE BENCHBOARD
5-A1	1H13-P877	DIESEL GENERATOR BENCHBOARD
7-A1	1H13-P881	DIV. 1 & DIV. 3 CNTMT & DW ISOLATION VALVE CONTROL PANEL
7-A2	1H13-P882	DIVISION 2 CONTAINMENT AND DRYWELL ISOLATION VALVE CONTROL PANEL
11-A4	1H13-P883	POST ACCIDENT MONITORING RECORDER PANEL-DIVISION 1 & 2
16-A1	1H13-P884	DIV. 1 POST ACCIDENT RADIATION MONITORING PANEL
16-A2	1H13-P885	DIV. 2 POST ACCIDENT RADIATION MONITORING PANEL
12	1H13-P895	UNIT SUPERVISOR'S CONSOLE
10-A1	1H13-P902	COMMON AIRBORNE RADIATION MONITORING PANEL
8-A1	1H13-P904	COM HEATING VENTILATION AND AIR CONDITIONING CON PNL
10-A2	1H13-P906	COMMON PROCESS & AREA RADIATION MONITORING PANEL
10-A3	1H13-P907	COMMON RADIATION MONITORING RECORDER PANEL
9	1H13-P969	COMMON ANALOG LOOP INSTR AND AUXILIARY RELAY PANEL
8-A2	1H13-P970	COMMON LONG RESPONSE CONTROL PANEL
21	1H13-PEI	PLANT EMERGENCY INST.

CEI FURNISHED EQUIPMENT SCHEDULE

UNIT NO.	CONTROL RM MPL NO.	DESCRIPTION
14-A2*	1C91-P701	PROCESS COMPUTER ALARM TYPER
	1C91-S600A	CORE PERFORMANCE VIDEO TERMINAL
14-A5*	1C91-S600B	CORE PERFORMANCE LOGGING TYPER
14-A4*	1C91-S600C	CORE PERFORMANCE HARDCOPY TERMINAL
	1C95-P620	ERIS RTAD PRINTER
	1C95-P637	ERIS TERMINAL
	1C95-P638	ERIS PRINTER/PLOTTER
	1H13-P613	PROCESS INSTRUMENTATION PANEL
	1H13-P618	DIVISION 2 AUXILIARY RELAY PANEL (RHR "B" & "C")
	1H13-P621	DIVISION 1 AUXILIARY RELAY PANEL (RCIC)
	1H13-P622	DIVISION 2 AUXILIARY RELAY PANEL (INBOARD VALVES)
	1H13-P623	DIVISION 1 AUXILIARY RELAY PANEL (OUTBOARD VALVES)
	1H13-P625	DIVISION 3 AUXILIARY RELAY PANEL (HPCS)
	1H13-P628	DIVISION 1 AUXILIARY RELAY PANEL (ADS CHANNEL "A")
	1H13-P629	DIVISION 1 AUXILIARY RELAY PANEL (LPCS & RHR "A")
	1H13-P631	DIVISION 2 AUXILIARY RELAY PANEL (ADS CHANNEL "B")
	1H13-P652	CONTROL ROD POSITION PANEL
	1H13-P654	DIV. 2 AUXILIARY RELAY PANEL (MSIV LEAK CONTROL SYS.)
	1H13-P655	DIV. 1 AUXILIARY RELAY PANEL (MSIV LEAK CONTROL SYS.)
	1H13-P669	DIV. 1 NEUTRON POWER & RAD INSTRUMENTATION PANEL
	1H13-P670	DIV. 2 NEUTRON POWER AND RAD INSTRUMENTATION PANEL
	1H13-P671	DIV. 3 NEUTRON POWER AND RAD INSTRUMENTATION PANEL
	1H13-P691	DIV. 1 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
	1H13-P692	DIV. 2 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
	1H13-P693	DIV. 3 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
	1H13-P694	DIV. 4 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
24	1H13-P805	OPERATOR'S CONSOLE
	1H13-P873	DIVISION 3 AUXILIARY RELAY PANEL
	1H13-P874	TYPER CONSOLE
	1H13-P5001	CONTROL ROOM COMMUNICATIONS PANEL
14-A1*	1R61-J230	SEQUENCE OF EVENTS TERMINET PRINTER
23		ERIS CRT AND KEYBOARD

* DENOTES CEI FURNISHED EQUIPMENT SHIPPED TO SINGER

NOTES:

1. WHERE OVERALL PANEL SIZE IS GREATER THAN THE SIZE OF THE PANEL BASE, ////// REPRESENTS THE AREA OF THE PANEL BASE.
2. THE SPACE OCCUPIED BY 1H13-P610 & P640 IN THE CONTROL ROOM WILL BE OCCUPIED BY 1H13-PEI IN THE SIMULATOR ROOM.
3. ◯ PANELS ARE FUNCTIONALLY SIMULATED

ARCHITECTURAL
TRAINING CENTER/EOF
SIMULATOR ROOM EQUIPMENT LAYOUT

SINGER FURNISHED EQUIPMENT SCHEDULE

UNIT NO	CONTROL RM MPL NO.	DESCRIPTION
6-A1	1C22-P001	REDUNDANT REACTIVITY CONTROL PANEL
6-A2	1C22-P002	REDUNDANT REACTIVITY CONTROL PANEL
14-A3	1C91-P637	PROCESS COMPUTER LINE PRINTER
15-A5	1H13-P600	RADIATION MONITORING RECORDER PANEL
5-A2	1H13-P601	EMERGENCY CORE COOLING SYSTEMS BENCHBOARD
15-A4	1H13-P604	PROCESS RADIATION MONITORING PANEL
15-A1	1H13-P614	NSSS RECORDER PANEL
16-A4	1H13-P619	JET PUMP MONITORING PANEL
20	1H13-P632	LEAK DETECTION PANEL-DIVISION 1
16-A5	1H13-P634	RECIRCULATION FLOW CONTROL INSTRUMENTATION PANEL
19	1H13-P642	LEAK DETECTION PANEL-DIVISION 2
17-A1	1H13-P651	CONTROL ROD POSITION PANEL
17-A2	1H13-P653	CONTROL ROD DRIVE CONTROL INSTRUMENTATION PANEL
18-A1	1H13-P672	NEUTRON/PROCESS RAD MON NON DIV
18-A2	1H13-P672	PROCESS RAD MON SRM & IRM CH. D/H

CEI FURNISHED EQUIPMENT SCHEDULE

UNIT NO.	CONTROL RM MPL NO.	DESCRIPTION
14-A2*	1C91-P701	PROCESS COMPUTER ALARM TYPER
	1C91-S600A	CORE PERFORMANCE VIDEO TERMINAL
14-A5*	1C91-S600B	CORE PERFORMANCE LOGGING TYPER
14-A4*	1C91-S600C	CORE PERFORMANCE HARDCOPY TERMINAL
	1C95-P620	ERIS RTAD PRINTER
	1C95-P637	ERIS TERMINAL
	1C95-P638	ERIS PRINTER/PLOTTER
	1H13-P613	PROCESS INSTRUMENTATION PANEL
	1H13-P618	DIVISION 2 AUXILIARY RELAY PANEL (RHR "B" & "C")
	1H13-P621	DIVISION 1 AUXILIARY RELAY PANEL (RCIC)
	1H13-P622	DIVISION 2 AUXILIARY RELAY PANEL (INBOARD VALVES)
	1H13-P623	DIVISION 1 AUXILIARY RELAY PANEL (OUTBOARD VALVES)
	1H13-P625	DIVISION 3 AUXILIARY RELAY PANEL (HPCS)
	1H13-P628	DIVISION 1 AUXILIARY RELAY PANEL (ADS CHANNEL "A")
	1H13-P629	DIVISION 1 AUXILIARY RELAY PANEL (LPCS & RHR "A")

18-A3	1H13-P672	[illegible]
1	1H13-P680	UNIT CONTROL CONSOLE
2	1H13-P680	UNIT CONTROL CONSOLE
3	1H13-P680	UNIT CONTROL CONSOLE
11-A3	1H13-P800	HEATING, VENTILATION & AIR CONDITIONING CONTROL PANEL
15-A2	1H13-P803	AREA RADIATION MONITORING PANEL
15-A3	1H13-P804	AIRBORNE RADIATION MONITORING PANEL
13-A2	1H13-P807	MAIN TRANSFORMER RELAY PANEL
13-A1	1H13-P808	GENERATOR RELAY PANEL
13-A4	1H13-P809	START-UP TRANSFORMER RELAY PANEL
13-A3	1H13-P810	UNIT AUXILIARY TRANSFORMER RELAY PANEL
13-A5	1H13-P811	ELECTRICAL RECORDER PANEL
11-A1	1H13-P823	TURBINE SUPERVISORY RECORDER PANEL
11-A2	1H13-P842	BALANCE OF PLANT RECORDER PANEL
16-A3	1H13-P845	OFF-GAS CONTROL PAN
4	1H13-P87J	LONG RESPONSE BENCHBOARD
5-A1	1H13-P877	DIESEL GENERATOR BENCHBOARD
7-C1	1H13-P881	DIV. 1 & DIV. 3 CNTMT & DW ISOLATION VALVE CONTROL PANEL
7-A2	1H13-P882	DIVISION 2 CONTAINMENT AND DRYWELL ISOLATION VALVE CONTROL PANEL
11-A4	1H13-P883	POST ACCIDENT MONITORING RECORDER PANEL-DIVISION 1 & 2
16-A1	1H13-P884	DIV 1 POST ACCIDENT RADIATION MONITORING PANEL
16-A2	1H13-P885	DIV.2 POST ACCIDENT RADIATION MONITORING PANEL
12	1H13-P895	UNIT SUPERVISOR'S CONSOLE
10-A1	1H13-P902	COMMON AIRBORNE RADIATION MONITORING PANEL
8-A1	1H13-P904	COM HEATING, VENTILATION AND AIR CONDITIONING COM PNL
10-A2	1H13-P906	COMMON PROCESS & AREA RADIATION MONITORING PANEL
10-A3	1H13-P907	COMMON RADIATION MONITORING RECORDER PANEL
9	1H13-P969	COMMON ANALOG LOOP INSTM AND AUXILIARY RELAY PANEL
8-A2	1H13-P970	COMMON LONG RESPONSE CONTROL PANEL
22	1H13-PEI	PLANT EMERGENCY INST.

	1H13-P63[illegible]	DIVISION 2 AUXILIARY RELAY PANEL [illegible] CHANNEL B [illegible]
	1H13-P[illegible]	CONTROL ROD POSITION PANEL
	1H[illegible]4	DIV.2 AUXILIARY RELAY PANEL (MSIV LEAK CONTROL SYS.)
	1H13-P655	DIV.1 AUXILIARY RELAY PANEL (MSIV LEAK CONTROL SYS.)
	1H13-P669	DIV.1 NEUTRON POWER & RAD INSTRUMENTATION PANEL
	1H13-P670	DIV.2 NEUTRON POWER AND RAD INSTRUMENTATION PANEL
	1H13-P671	DIV.3 NEUTRON POWER AND RAD INSTRUMENTATION PANEL
	1H13-P691	DIV.1 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
	1H13-P692	DIV.2 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
	1H13-P693	DIV.3 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
	1H13-P694	DIV.4 RPS INSTRUMENTATION AND AUXILIARY RELAY PANEL
24	1H13-P805	OPERATOR'S CONSOLE
	1H13-P873	DIVISION 3 AUXILIARY RELAY PANEL
	1H13-P871	TYPER CONSOLE
	1H13-P5001	CONTROL ROOM COMMUNICATIONS PANEL
14-A1*	1H61-J230	SEQUENCE OF EVENTS TERMINET PRINTER
23		ERIS CRT AND KEYBOARD

* DENOTES CEI FURNISHED EQUIPMENT SHIPPED TO SINGER

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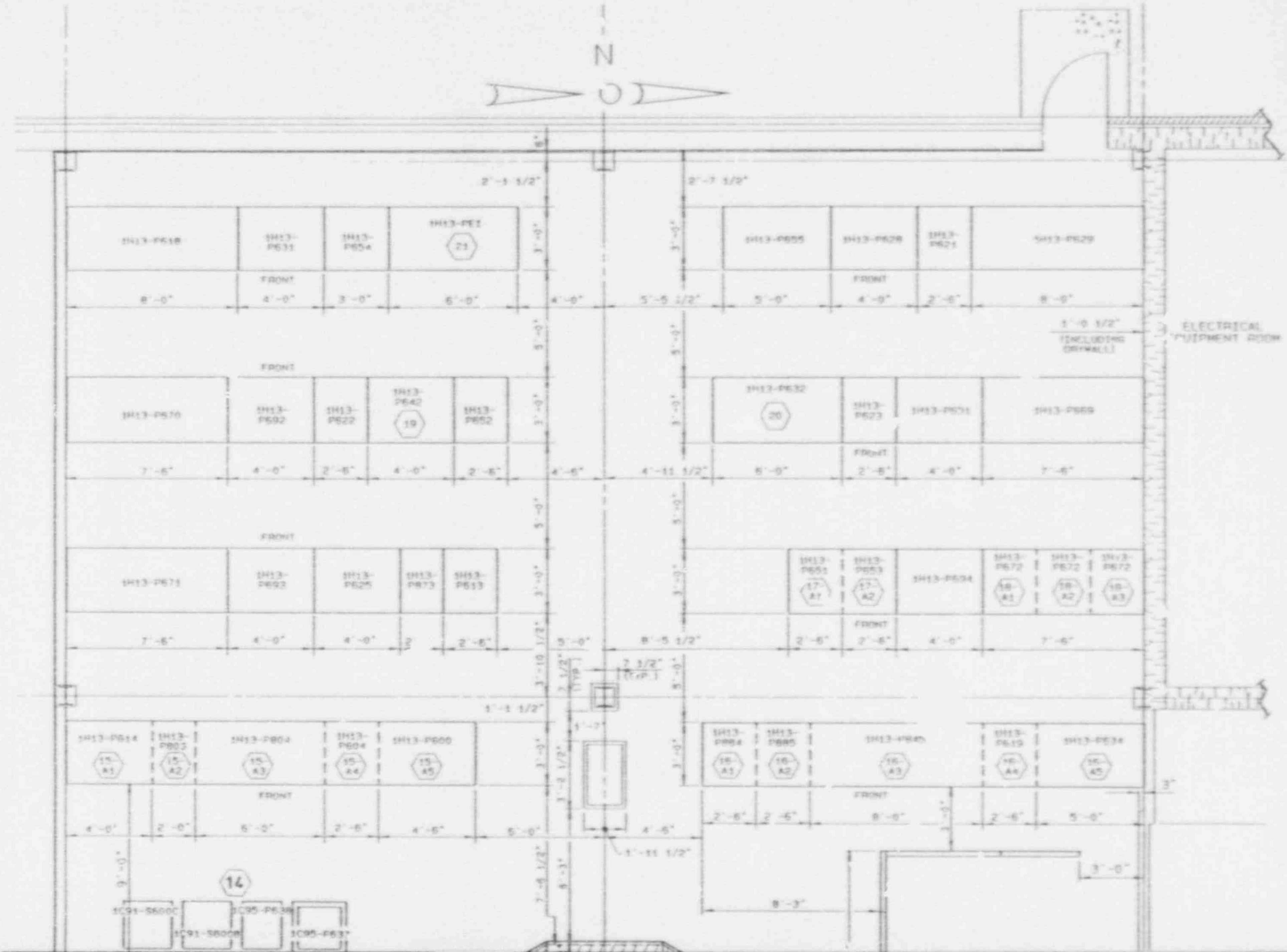
NOTES:

1. WHERE OVERALL PANEL SIZE IS GREATER THAN THE SIZE OF THE PANEL BASE, ////////// REPRESENTS THE AREA OF THE PANEL BASE.
2. THE SPACE OCCUPIED BY 1H13-P610 & P640 IN THE CONTROL ROOM WILL BE OCCUPIED BY 1H13-PEI IN THE SIMULATOR ROOM.
3. ⬡ PANELS ARE FUNCTIONALLY SIMULATED

ARCHITECTURAL
TRAINING CENTER/EOF
SIMULATOR ROOM EQUIPMENT LAYOUT

SI APERTURE CARD

Also Available On Aperture Card


N
ELECTRICAL EQUIPMENT ROOM
1H13-P610
1H13-P631
1H13-P654
1H13-PEI
21
1H13-P655
1H13-P628
1H13-P621
1H13-P629
FRONT
1H13-P670
1H13-P692
1H13-P622
1H13-P642
19
1H13-P652
1H13-P632
20
1H13-P623
1H13-P631
1H13-P669
1H13-P671
1H13-P693
1H13-P625
1H13-P873
1H13-P613
1H13-P651
1H13-P653
1H13-P634
1H13-P672
1H13-P614
1H13-P803
1H13-P804
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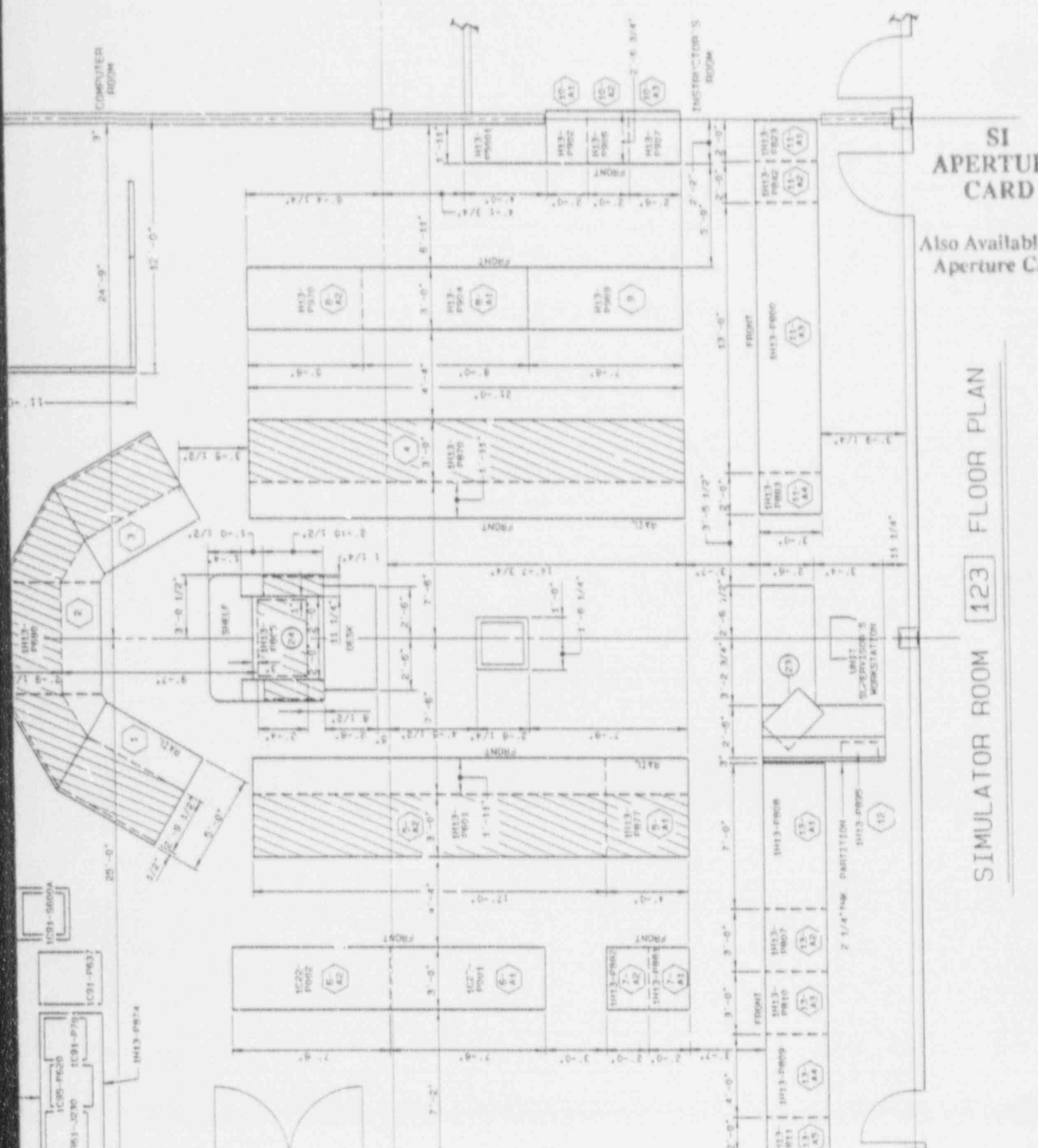
1H13-P884
1H13-P885
1H13-P845
1H13-P619
1H13-P634
14
1C91-S600C
1C91-S600B
1C95-P638
1C95-P637
1'-0 1/2" (INCLUDING DRYWALL)



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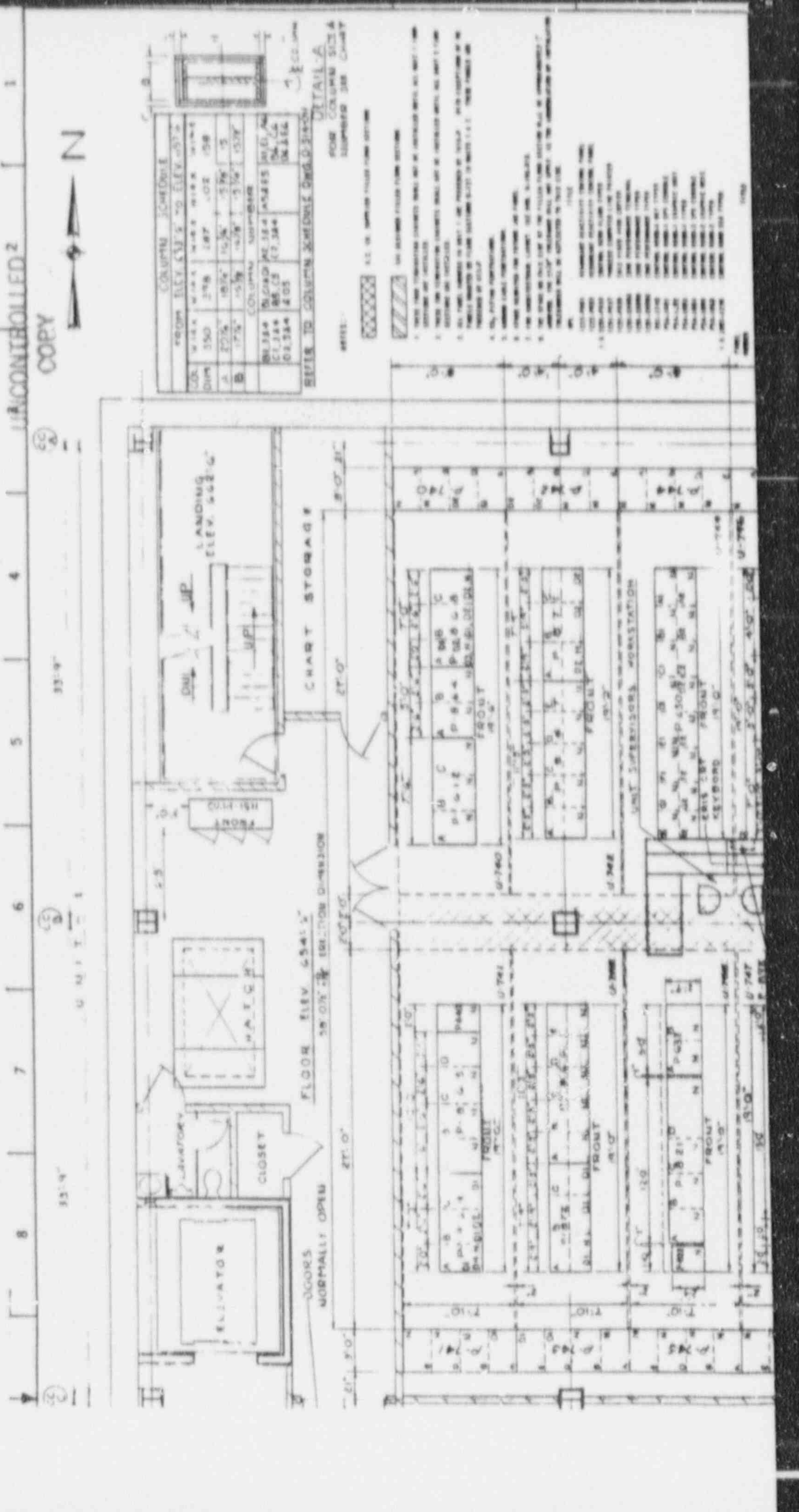

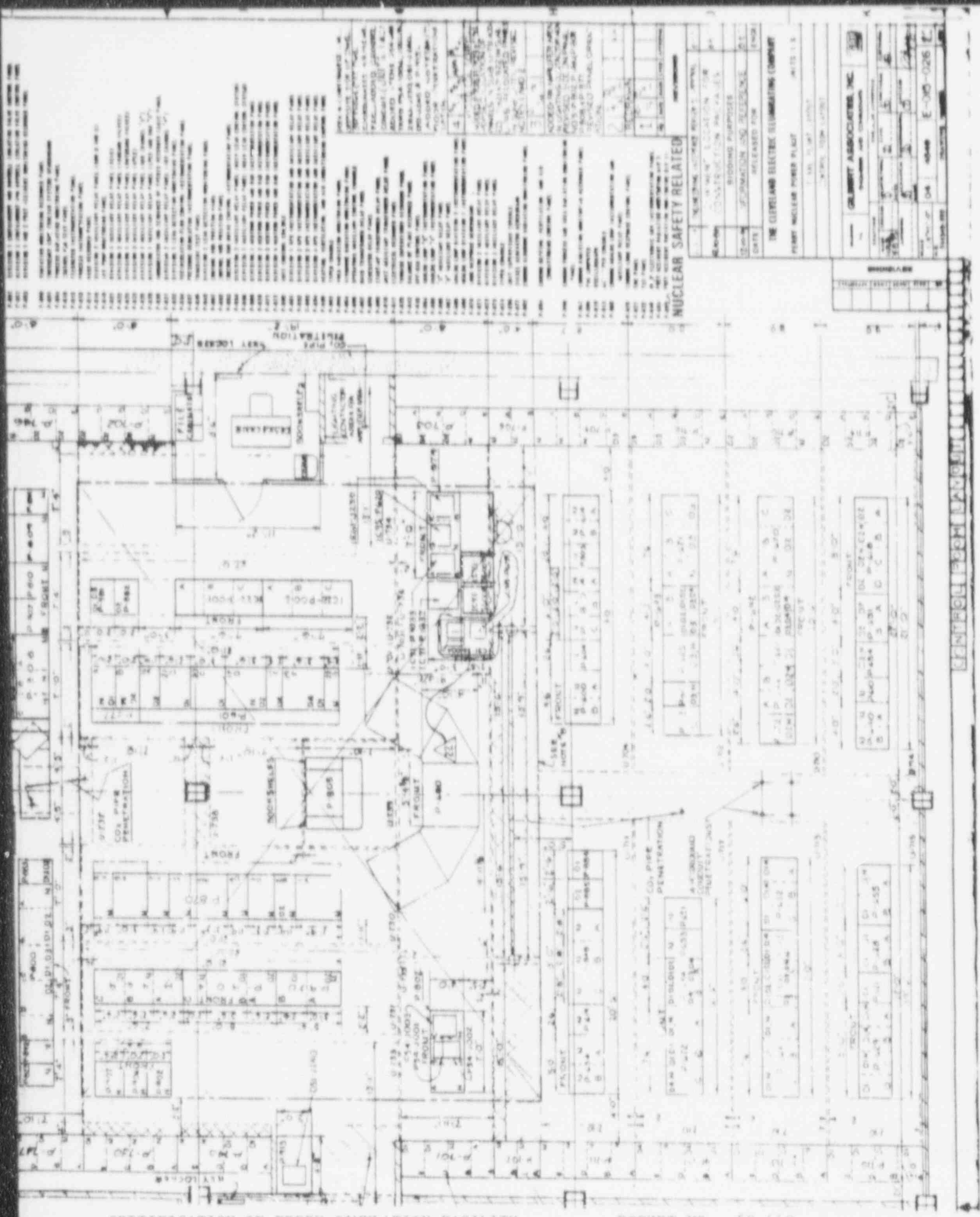

9107020106-03

Attachment 3 - Generic Malfunction List

Each malfunction identified below is verified in one or more of the tests listed in TAB C Parts 4, 5A, and 5B. An asterisk (*) in the "Test #" column indicates that the malfunction is included in more tests than the one shown. For more complete information, refer to the "Malfunctions Tested" section of each test abstract included in TAB C to determine the specific malfunctions utilized in any particular Certification Test.

Malf. ID	Test #	Malfunction Description
AD01	T.4.5.3.22*	CYCLING SAFETY RELIEF VALVE
AN01	T.4.5.1.02	ANNUNCIATOR INPUT OPTICAL ISOLATOR FAILURE
AV02	T.4.5.3.27*	AIR OPERATED VALVE FAILS CLOSED
AV03	T.4.5.3.25	AIR OPERATED VALVE FAILS AS IS
BS01	T.4.5.3.28*	BISTABLE FAILS TO TRIP
BS02	T.4.5.3.27*	BISTABLE SPURIOUS TRIP
CB01	T.4.5.3.08*	SPURIOUS BREAKER TRIP
CB03	T.4.5.3.23	BREAKER AUTO TRIP LOGIC FAILURE
CB04	T.4.5.3.34	BREAKER AUTO CLOSE LOGIC FAILURE
CB05	T.4.5.3.27*	BREAKER FAILS IN CURRENT POSITION (MECHANICAL SEIZURE)
CB06	T.4.5.3.22*	BREAKER FAILS IN CURRENT POSITION (LOSS OF CONTROL POWER)
CB07	T.4.5.3.24	BREAKER FAILS TO CLOSE
CN01	T.4.5.3.03*	CONTROLLER AUTO/MANUAL FAILURE
CN02	T.4.5.3.30	CONTROLLER AUTO FAILURE
CP01	T.4.5.3.16	PUMP SHAFT BREAKS
CP02	T.4.5.3.14*	PUMP SHAFT SEIZES
CP03	T.4.5.3.25*	PUMP HEAD LOSS (FLOW DEGRADATION)
CU03	T.4.5.1.04	RWCU SYSTEM PIPE BREAK OUTSIDE CONTAINMENT (STEAM TUNNEL)
DG03	T.4.5.3.26*	DIESEL GEN SPEED GOVERNOR FAILS
DG06	T.4.5.3.26*	FUEL OIL DAY TANK LEAK
ED05	T.4.5.3.26	LOSS OF 4.16 KV BUS
ED06	T.4.5.1.06	LOSS OF 480V BUS
ED07	T.4.5.3.22	LOSS OF 120V BUS
ED09	T.4.5.1.06	LOSS OF 125V DC BUS
ED16	T.4.5.3.34	LOSS OF 480V MOTOR CONTROL CENTER
ED17	T.4.5.1.06	LOSS OF 125V DC DISTRIBUTION PANEL
EG01	T.4.5.1.07	MAIN GENERATOR LOCKOUT RELAY TRIP
FW02	T.4.5.1.09	FEEDWATER SYSTEM PIPE BREAK INSIDE DRYWELL
FW03	T.4.5.1.09	FEEDWATER SYSTEM PIPE BREAK OUTSIDE CONTAINMENT
FW04	T.4.5.3.29	FEED PUMP LOGIC FAILURE
FW08	T.4.5.3.28	FEEDWATER PUMP LOSS OF LUBRICATING OIL
HX02	T.4.5.3.25	HEAT EXCHANGER TUBE LEAK

Attachment 3 - Generic Malfunction List

Malf. ID	Test #	Malfunction Description
IA01	T.4.5.1.12	AIR RECEIVER LEAK
IA02	T.4.5.1.12	INSTRUMENT AIR LINE LEAK
MC01	T.4.5.1.15*	CONDENSER AIR INLEAKAGE
MC02	T.4.5.3.29	CONDENSER TUBE LEAK
MS03	T.4.5.3.04	MAIN STEAM ISOLATION VALVE CLOSURE TIME VARIANCE
MS11	T.4.5.1.16	STEAM SEAL HEADER PRESSURE REGULATOR FAILURE
MV01	T.4.5.3.31	MOTOR OPERATED VALVE FAIL AS IS (LOSS OF CONTROL POWER)
MV02	T.4.5.3.30	MOTOR OPERATED VALVE SPURIOUS VALVE OPENING
MV03	T.4.5.3.01*	MOTOR OPERATED VALVE SPURIOUS VALVE CLOSURE
MV04	T.4.5.3.37	MOTOR OPERATED VALVE FAILURE OF AUTO OPEN CIRCUIT
MV06	T.4.5.3.12*	MOTOR OPERATED VALVE FAIL AS IS (MECHANICAL BINDING)
NM01	T.4.5.1.17	SOURCE RANGE MONITOR DETECTOR (PRE-AMP) FAILURE
NM02	T.4.5.1.17	INTERMEDIATE RANGE MONITOR DETECTOR (PRE-AMP) FAILURE
NM03	T.4.5.1.17	LOCAL POWER RANGE MONITOR DETECTOR FAILURE
NM04	T.4.5.3.22*	AVERAGE POWER RANGE MONITOR OUTPUT FAILURE
NM10	T.4.5.1.17	NEUTRON MONITORING DETECTOR DRIVE STUCK
OG03	T.4.5.1.18	OFF GAS SYSTEM LEAK UPSTREAM OF ADSORBERS
OG04	T.4.5.1.18	OFF GAS SYSTEM LEAK DOWNSTREAM OF ADSORBERS
PC01	T.4.5.3.25	INCREASED DRYWELL/CONTAINMENT BYPASS LEAKAGE
PC04	T.4.5.3.40	SUPPRESSION POOL LEAK
PT01	T.4.5.3.04*	PROCESS TRANSMITTER VARIABLE FAILURE
PT03	T.4.5.3.38	PROCESS TRANSMITTER VARIABLE OUTPUT CLAMP
RC04	T.4.5.3.26	REACTOR CORE ISOLATION COOLING GOVERNOR VALVE FAILURE
RD01	T.4.5.3.18*	STUCK CONTROL ROD
RD02	T.4.5.3.18*	UNCOUPLED CONTROL ROD
RD03	T.4.5.1.21	CONTROL ROD DRIFT - IN
RD04	T.4.5.1.21	CONTROL ROD DRIFT - OUT
RD05	T.4.5.3.23*	CONTROL ROD ACCUMULATOR FAULT
RD12	T.4.5.3.36	SCRAM OUTLET VALVE LEAK
RD15	T.4.5.3.20*	ANTICIPATED TRANSIENT WITHOUT SCRAM
RD17	T.4.5.3.22*	LOSS OF CONTROL ROD DRIVE PUMP LUBE OIL
RD18	T.4.5.3.36*	SCRAM DISCHARGE VOLUME DRAIN BLOCKAGE
RH02	T.4.5.1.22*	RESIDUAL HEAT REMOVAL SYSTEM PIPE BREAK
RP01	T.4.5.3.23*	ELECTRICAL PROTECTION ASSEMBLY TRIP
RP02	T.4.5.1.23	INADVERTENT INITIATION OF ALTERNATE ROD INSERTION
RP03	T.4.5.3.23*	FAILURE OF ALTERNATE ROD INJECTION TO INITIATE
RP04	T.4.5.3.23	INADVERTENT REDUNDANT REACTIVITY CONTROL SYSTEM FEEDWATER RUNBACK, REACTOR RECIRCULATION DOWNSHIFT, LOW FREQUENCY MOTOR GENERATOR TRIP
RV02	T.4.5.3.33	RELIEF VALVE STUCK
RV03	T.4.5.3.23*	RELIEF VALVE FAILS OPEN

Attachment 3 - Generic Malfunction List

Malf. ID	Test #	Malfunction Description
RY01	T.4.5.3.24	RELAY FAILS DE-ENERGIZED
RY02	T.4.5.3.22*	RELAY FAILS AS IS
SL05	T.4.5.3.26*	STANDBY LIQUID CONTROL INJECTION PIPING LEAK
SW01	T.4.5.1.25	NUCLEAR CLOSED COOLING SYSTEM PROCESS PIPING LEAKAGE
SW02	T.4.5.1.25	SERVICE WATER SYSTEM PROCESS PIPING LEAKAGE
SW03	T.4.5.3.31	TURBINE BUILDING CLOSED COOLING SYSTEM PROCESS PIPING LEAKAGE
SW07	T.4.5.3.29	LOSS OF COMPONENT COOLING - TURBINE BUILDING CLOSED COOLING
TC04	T.4.5.3.09*	BYPASS VALVE FAILURE
TF01	T.4.5.3.11*	LOSS OF TRANSFORMER
TH01	T.4.5.1.27*	RECIRC LOOP RUPTURE (DESIGN BASIS ACCIDENT LOCA)
TH02	T.4.5.3.25	RECIRC LOOP PIPING BREAK
TH14	T.4.5.3.33*	RECIRC FLOW CONTROL VALVE HYDRAULIC POWER UNIT OIL HIGH TEMP
TH15	T.4.5.3.26	GROSS FUEL FAILURE
TH19	T.4.5.1.27	REACTOR PRESSURE VESSEL LEVEL INST REFERENCE LEG BREAK
TH20	T.4.5.1.27	REACTOR PRESSURE VESSEL LEVEL INST VARIABLE LEG BREAK
TH21	T.4.5.1.27	POWER/FLOW INSTABILITIES (IEB 88-07 SUPPLEMENT 1)
TH26	T.4.4.2.10*	MAIN STEAM LINE RUPTURE INSIDE DRYWELL
TH27	T.4.5.1.27	MAIN STEAM LINE RUPTURE IN STEAM TUNNEL
TH28	T.4.5.3.25	MAIN STEAM LINE BREAK INSIDE GUARD PIPE
TU01	T.4.5.1.28	MAIN SHAFT OIL PUMP DEGRADATION

SIMULATOR DESIGN DATA

Data used to design the Perry Plant simulator is available through review of baseline information in the Configuration Management System (CMS). Types of data include (1) drawings and their change documents, i.e., Configuration Update Notices (CUNs) and Drawing Change Notices (DCNs), (2) Design Change Packages (DCPs), (3) Setpoint List information, (4) vendor and technical manuals, (5) miscellaneous plant data requested through formal "Data Requests," (6) Photos taken of Control Room panels for comparison purposes, and (7) Perry Plant System Operating Instructions. This information is available either in whole by logging on to the CMS computer system and viewing it or by reference to specific hardcopy documentation which is available in binders or on microfilm.

The design of the simulator reflects the configuration of the referenced plant as of July 1, 1989 (the date of design freeze) with the exception that some later (updated) design information was incorporated when determined necessary and practical. The Configuration Management System provides the actual date of baselining for each design input item.

SIMULATOR DISCREPANCY RESOLUTION AND UPGRADING

A. Identification, Correcting, and Testing of Discrepancies

The Perry Training Section, through a Certification Administrator, maintains the Perry Plant Simulator by interfacing with a matrix organization which consists of personnel from the software engineering, hardware engineering, instrumentation and controls, and operator training groups. Simulator discrepancies are identified in several ways: by the Certification Administrator or Software Engineer through direct review of plant change documents shortly after completion of the changes in the plant; by the Software Engineer and/or Hardware Engineer during the plant change cost estimate process when the simulator group provides simulator change cost estimates related to plant design changes; by operator trainees or instructors who make an observation and feed it back to the Certification Administrator using a Student Instructor Feedback Report; or by anyone in the matrix organization observing a problem while making modifications or conducting daily, annual, or periodic testing.

For each discrepancy a Simulator Discrepancy Package (SDP) is initiated in accordance with instruction OM14: TMA-4206 and is processed through analysis, implementation, and testing phases by the Certification Administrator who also maintains appropriate tracking information in the Update Data Base portion of the computerized Configuration Management System. Instructions are available on site which describe this process in more detail.

B. Tracking of Plant Design Changes

Shortly after completing a change in the plant and declaring the modified item or system operable, the Perry Training Section is sent a copy of the change documents for review to determine any impact on training. The Certification Administrator completes a Training Significance Review (TSR) in accordance with instruction OM14: TMA-4204 and, for those changes affecting the Simulator, initiates a Simulator Discrepancy Package and enters the SDP number into the Update Data Base for tracking purposes. The default due date for completing the change in the simulator is one year from the date the TSR was completed, however, the Operator Training Unit may assign an earlier date for higher priority changes.

On an annual basis prior to the Simulator Certification anniversary date, a comparison of the current revision of design drawings, CUNs, and DCNs will be made with the revision listed in the simulator Configuration Management System. For any discrepancy identified, the Certification Administrator will determine whether the item is already being tracked on an SDP initiated through the TSR process or whether a new SDP is needed. This annual review process ensures that plant changes are implemented within the simulator preferably within one year of the associated TSR review date but definitely no later than the next annual review date or two years from the date of the change in the plant, whichever is earlier.

TAB B EXCEPTIONS TO ANS 3.5

Part 1. Permanent Exceptions

Part 2. Temporary Exceptions (Discrepancies)

A. Exceptions Resulting from Filing Form 474 Prior to Simulator Delivery and Installation

B. Exceptions Related to Work Identified after Design Freezes were Established

Part 3. Schedule for Correcting Discrepancies

A. Work to be Completed Prior to Declaring the Simulator "Ready for Training"

B. Work to be Completed Before June 28, 1992

PERMANENT EXCEPTIONS TO ANSI/ANS 3.5

Permanent Exceptions to ANSI/ANS 3.5 are listed by "PE" number and identify the applicable ANSI/ANS 3.5 section or paragraph to which the exception is being taken. Identified differences between the plant and simulator were in the area of control room environmental differences only. These differences, augmented with compensatory training given to operator trainees where necessary, have been determined by the Simulator Review Board to have minimal impact on training and are, therefore, acceptable exceptions. The Training Impact Evaluations (TIEs) referenced for each exception provide specific analysis and review documentation and are available at the Perry site.

PE 001

ANS 3.5 Section: 3.2.1 Degree of Panel Simulation

Component Affected: Control Room Panels 1H13-P613, P618, P621, P622, P623, P625, P628, P629, P631, P652, P654, P655, P669, P670, P671, P691, P692, P693, P694, and P873 (Ref. Control Room Layout, Tab A, Part 1, Attachment 2).

Description: Panels are physically there but are not functionally included in the scope of the new simulator. (Ref. Simulator Room 123 Floor Plan, Tab A, Part 1, Attachment 1).

Related TIEs: 91-016, 91-017, 91-018, 91-019, 91-020, 91-021, 91-022, 91-023, 91-025, 91-028, 91-029, 91-030, 91-031, 91-032, 91-033, 91-034, 91-035, 91-036, 91-037, 91-049

Testing/Training Impact: With the exception of off-normal and emergency controls (addressed by PE 002), the controls and indications on these panels that are manipulated or monitored by operators have minor impact and need not be simulated for one or more of the following reasons: a) panels for redundant divisions are simulated, b) operator actions are routine (performed daily), c) classroom training or OJT adequately covers these controls and indications, d) panel has no impact on any training scenario.

PE 002

ANS 3.5 Section: 3.2.2 Controls on Panels

Component Affected: Panels 1H13-P691, 692, 693, 694, 625, 618, 629, 631, 640, 866, 871, 872, and 628 with the switches, relays, fuses, and terminal boards that are operated during Plant Emergency Instructions (PEI's) and Off Normal Instructions (ONI's).

Description: These 13 panels have been incorporated into one panel in the simulator (Unit 21, 1H13PEI, installed in the approximate location of control room panel 1H13P640).

PERMANENT EXCEPTIONS TO ANSI/ANS 3.5

(PE 002 continued)

Related TIEs: 91-005

Testing/Training Impact: These components are manipulated more in the simulator (during training) than in the plant, therefore, the operator's knowledge of their actual plant location is important. Job Performance Measures (JPM's) have been or will be written to provide training on operation and actual location of the devices operated on the above panels. These JPM's are a required part of the training program to ensure that operators know the actual location of each device.

PE 003 ANS 3.5 Section: 3.2.1 Degree of Panel Simulation

Component Affected: Unit 1 Control Room Panels 1H13-P610, P612, P630, P637, P640, P821, P822, P840, P864, P865, P866, P867, P868, P869, P871, P872, P913 (Ref. Control Room Layout, Tab A, Part 1, Attachment 2)

Description: Panels are not included in the scope of the new simulator. (Ref. Simulator Room 123 Floor Plan, Tab A, Part 1, Attachment 1)

Related TIEs: 91-014, 91-015, 91-024, 91-026, 91-027, 91-038, 91-039, 91-040, 91-041, 91-042, 91-043, 91-044, 91-045, 91-046, 91-047, 91-048, 91-052

Testing/Training Impact: The absence of these panels in the simulator has minor impact for one or more of the following reasons: a) operator actions are routine (performed daily) and are addressed by OJT, b) off-normal and emergency controls are simulated on the PEI panel (see PE 002), c) classroom training and/or Job Performance Measures (JPMs) adequately address the location and function of the panel, d) no external controls or indications exist on the panel.

PE 004 ANS 3.5 Section: 3.2.3 Control Room Environment

Component Affected: Ceiling differences in Unit 1 Control Room vs. Simulator Room.

Description: The Control Room upper and lower ceilings are 17' 6" and 10' 6" high, respectively, with 1'x1' hidden spline acoustic tile in the lower ceiling and a motorized light dimming control for the upper ceiling. The Simulator Room ceilings are 10' and 9' high, with 2'x4' lay-in acoustic tile in the lower ceiling and air diffuser slots in both. The Simulator lighting does not have a dimming control.

PERMANENT EXCEPTIONS TO ANSI/ANS 3.5

(PE 004 continued)

Related TIEs:	91-008, 91-009, 91-010, and 91-013
Testing/Training Impact:	There is no impact to training due to the differences in ceiling configuration. Area lighting levels can be controlled at the Simulator Room Unit Supervisor's workstation consistent with available lighting levels in the Control Room but by using different means. Emergency lighting levels at the Simulator Room control panels will be maintained within NUREG 0700 guidelines. Other visual differences in the ceilings, although noticeable upon inspection, have negligible impact to training since no operator interface with ceilings is required.
PE 005 ANS 3.5 Section:	3.2.3 Control Room Environment
Component Affected:	Structure and size of Unit 1 Control Room (EL 654'-6") vs. Simulator Room (TEC123).
Description:	The Simulator Room is smaller in size and different in structure from the Control Room in that 1) outside aisles to the north and south of Simulator Room back panels are missing, 2) exit and office doors are by photo mockup only, 3) some miscellaneous furniture cannot be included, and 4) an additional 15"x15" column exists in the Simulator Room.
Related TIEs:	91-004, 91-006, 91-011, and 91-012
Testing/Training Impact:	There is no impact or minor impact to training from the difference in environment. Where evaluations determined minor impact (i.e., with respect to missing aisles and the extra column), compensatory training has been or will be incorporated into the training program. The column causes negligible visibility impact since panel components were not designed to be distiguishable from the column area. The column and missing aisles cause minor impact in that a trainee may become accustomed to a slightly longer path to panels than required in the control room.

TEMPORARY EXCEPTIONS TO ANSI/ANS 3.5 (DISCREPANCIES)

The following lists of Temporary Exceptions (designated by "TE" numbers) represent all exceptions to ANSI/ANS 3.5 for which a schedule has been established to correct the discrepancies that created the exception.

These exceptions fall into one of two categories: A) those resulting from filing Form 474 prior to shipment of the Simulator from the vendor's facility; or B) those resulting from discrepancies identified after freezing design input to the new simulator on July 1, 1989. Followup to ensure closure of these exceptions is through the use of a Simulator Discrepancy Report (SDR) issued by the Simulator vendor, S3 Technologies, or through use of a Simulator Discrepancy Package (SDP) issued by CEI at Perry.

Once performance testing has been completed (reference: exception TE 001), the makeup of these lists will likely have changed. Therefore, a supplement will be made to this Form 474 submittal within one month after declaring the Simulator "Ready for Training" in order to update this portion of the certification report.

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TEMPORARY EXCEPTIONS TO ANSI/ANS 3.5 (DISCREPANCIES)

Section A

The exceptions in Section A (TE 001 through TE 004) represent the work to be done prior to declaring the Simulator "Ready for Training". They are the result of filing Form 474 prior to shipment of the simulator from the vendor's facility and installation/in-place testing of the simulator at Perry. These discrepancies will be handled on a high priority basis as soon as feasible but before declaring the simulator "Ready for Training."

TE 001

ANS 3.5 Section: 3. General Requirements; 5.4 Simulator Testing

Nature of Exception: Simulator Performance Testing has not been completed and/or major Simulator Discrepancy Reports (SDRs) remain open rendering the Simulator "not ready for training." The "Tests included in this Section" listing in each Part to Tab C indicates which tests are incomplete by displaying an asterisk and either "(N/C)" for those that have not been successfully run or a number for the number of open SDRs associated with completed tests which constitute exceptions to ANSI/ANS 3.5.

Anticipated Resolution: 07/20/91

Related SDPs/SDRs: SDRs - See "Tests included in this Section" at the beginning of each Part in Tab C.

TE 002

ANS 3.5 Section: 3. General Requirements; 5.4 Simulator Testing

Nature of Exception: The Simulator is not yet installed or tested in its permanent location and, therefore, cannot function acceptably as a training device/examination tool. Remaining work to be conducted after installation includes On-Site Reverification Testing (including re-run of ANSI/ANS 3.5 App. B transients; ERIS Interface Test; and adjustment and operational verification of peripheral interfaces such as communication and lighting systems.

Anticipated Resolution: 08/31/91

Related SDPs/SDRs: SDPs 91-1060 and 91-1062

TEMPORARY EXCEPTIONS TO ANSI/ANS 3.5 (DISCREPANCIES)

TE 003

ANS 3.5 Section: 3.2.2 Controls on Panels

Nature of Exception: Various hardware items currently in use on the old simulator must be reinstalled on the new simulator after simulator installation at Perry. These items include pushbutton switches and annunciator windows.

Anticipated Resolution: 08/31/91

Related SDPs/SDRs: SDPs 91-1010, 91-1024, and 91-1071

TE 004

ANS 3.5 Section: 3.3 Systems to be Simulated and Degree of Completeness

Nature of Exception: Hardware and Software not within the vendor's scope of work but required for proper simulation must be installed after delivery of the new simulator.

Anticipated Resolution: 08/31/91

Related SDPs/SDRs: SDPs 90-1012, 90-1031, 91-1014, 91-1015, 91-1016, 91-1023, 91-1039, 91-1051, 91-1057, 91-1058, and 91-1059

TEMPORARY EXCEPTIONS TO ANSI/ANS 3.5 (DISCREPANCIES)

Section B

The exceptions in Section B (TE 005 through TE 008) resulted from discrepancies identified after design freezes were established. These discrepancies will be resolved prior to June 28, 1992 using the schedule shown in Part 3 to Tab B.

TE 005	
ANS 3.5 Section:	3.1 Simulator Capabilities
Nature of Exception:	Setpoint changes completed in the plant between 11/07/89 and 04/10/91 must be implemented in the simulator.
Anticipated Resolution:	See Tab B, Part 3
Related SDPs/SDRs:	SDPs 90-1045 through 90-1079, 90-1086, 90-1087, 90-1090, 90-1091, 91-1011, 91-1017 through 91-1022, 91-1027 through 91-1034, 91-1047 through 91-1050, 91-1056, and 91-1063 through 91-1068

TE 006	
ANS 3.5 Section:	3.1 Simulator Capabilities
Nature of Exception:	Various changes to software must be made to implement plant changes made between 12/08/89 and 05/15/91 and to modify computer systems used by Control Room operators.
Anticipated Resolution:	See Tab B, Part 3
Related SDPs/SDRs:	SDPs 90-1028, 90-1042, 90-1043, 90-1044, 90-1081, 90-1084, 90-1088, 90-1092, 90-1094, 90-1098, 91-1001, 91-1002, 91-1003, 91-1004, 91-1005, 91-1006, 91-1007, 91-1008, 91-1012, 91-1013, 91-1025, 91-1040, 91-1041, 91-1069, and 91-1070

TE 007	
ANS 3.5 Section:	3.2 Simulator Environment; 3.3 Systems to be Simulated and Degree of Completeness
Nature of Exception:	Various hardware changes with related software changes need to be made to incorporate changes made to the plant after 11/15/90.
Anticipated Resolution:	See Tab B, Part 3
Related SDPs/SDRs:	SDPs 90-1022, 90-1085, 91-1044, 91-1045, and 91-1046

TEMPORARY EXCEPTIONS TO ANSI/ANS 3.5 (DISCREPANCIES)

TE 008	ANS 3.5 Section:	3.2 Simulator Environment
	Nature of Exception:	Miscellaneous changes or additions to hardware need to be made to make labels, nameplates, tags, operator aides, panel hardware, and furniture consistent with those items in the Control Room.
	Anticipated Resolution:	See Tab B, Part 3
	Related SDPs/SDRs:	SDPs 90-1018, 90-1080, 90-1082, 90-1083, 90-1089, 90-1093, 90-1097, 91-1009, 91-1035, 91-1036, 91-1037, 91-1038, 91-1042, 91-1043, 91-1052, 91-1053, and 91-1061

SCHEDULE FOR CORRECTING DISCREPANCIES

The Schedule for Correcting Discrepancies provides dates for completing work related to Temporary Exceptions TE 002 through TE 008 (i.e., modifications not associated with performance testing at the Simulator vendor's facility). A supplement will be made to this Form 474 submittal within one month after declaring the Simulator "Ready for Training (RFT)" in order to update this schedule for any work related to resolving test discrepancies (i.e. Simulator Discrepancy Reports).

Section A of this schedule describes work which will be completed prior to declaring the simulator "Ready for Training." Each "scheduled completion" date is listed as 08/31/91 which is the anticipated RFT date.

Section B of this schedule describes work which will be completed after declaring the simulator "Ready for Training" but before June 28, 1992. This work is necessary to bring simulator data and hardware into conformance with plant data and hardware.

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Schedule for Correcting Discrepancies Tracked by SDP Number
Section A - To Complete Prior to RFT

SDP 90-1012 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Attach "Simulator Emergency Power Off" labels within 3 inches of emergency power off switches. Also install protective collars on each switch.

SDP 90-1031 | Initiating Document: DCP 86-0075/000/00 | Plant Date: 07/21/89 | Scheduled Completion: 08/31/91

Description: G33 - Change MPL G33F0041 from fail OPEN to fail CLOSE.

SDP 91-1010 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Replacement of Cutler Hammer, type E20 arm/depress pushbutton switches, located on 1H13P0601 and 1H13P0680. Hardware from the old simulator is used.

SDP 91-1014 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: ERIS Display 033, 'Hydrogen Concentration' requires an operational test. The previous simulation did not provide the variables necessary. Part of ERIS ATP.

SDP 91-1015 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Software modifications to support interface of the 32/77 ERIS computer to the Encore 2040's in the new simulator.

SDP 91-1016 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Back panel annunciators must be re-engraved to match human factors annunciator engraving standards. Current engravings use improper fonts.

SDP 91-1023 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Installation of Superphones and Plant Public Address on new simulator panels 1H13P0870, 1H13P0601, and 0H13P0969.

SDP 91-1024 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Replacement of simulator annunciator windows on 1H13P0680, 1H13P0870, 1H13P0877, and 1H13P0601, simulator units 1 through 5. Windows from the old simulator are used.

SDP 91-1039 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 08/31/91

Description: Install two Versatec multiplexers in 1H13P0874 (Unit 14) and associated cabling between basic control units in 1C22P0001 & 1C22P0002 (Unit 6) & V-80 Printer/Plotter.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section A - To Complete Prior to RFT

SDP 91-1051 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 08/31/91

Description: Provide a means for simulator communication equipment to operate in either "E-Plan Mode" (tied into plant comm. system) or "Training Mode" (isolate/simulate plant communication system).

SDP 91-1057 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 08/31/91

Description: Revise the ERIS valve position calculations for AC powered valves that use the Control Room valve status lights.

SDP 91-1058 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 08/31/91

Description: Update scales for meters 1N27R0411A/B, 1G43R0062A/B and 1C34R0608 per Human Factors initiated WO's 90-4328, 90-4325, 90-4326 and 90-4329.

SDP 91-1059 Initiating Document: DCP 88-0082/001/00 Plant Date: 07/15/89 Scheduled Completion: 08/31/91

Description: Reverse Bailey Controller Meter output so "Close" is on left of scale and "Open" is on right of scale.

SDP 91-1060 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 08/31/91

Description: After installation of the new simulator, adjust ceiling-mounted emergency light cans to illuminate appropriate areas of the control room.

SDP 91-1062 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 08/31/91

Description: S3 Technologies will be responsible for removal of the old simulator and installation of the new simulator. After power-up, post installation testing must be conducted.

SDP 91-1071 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 08/31/91

Description: Remove two black ERIS screen bezels from existing simulator panel 1H13P0680 and install on replacement simulator unit 2 (1H13P0680).

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 90-1018 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 09/15/91

Description: Attach MPL nameplates on proper equipment after delivery of new simulator per the latest copy of D-128-051, Computer Room and Instructor's Room Equipment Layout.

SDP 90-1022 Initiating Document: DCP 90-0142/001/00 Plant Date: 12/22/90 Scheduled Completion: 09/15/91

Description: Replace current R61 Sequence of Events Recorder CPU & LPU (RIS) and printer (Terminet) with a Beta Products SER and DEC LA-210 printer.

SDP 90-1028 Initiating Document: DCP 89-0259/000/00 Plant Date: 12/08/89 Scheduled Completion: 12/08/91

Description: E31 - Relocate RWCU pump room temperature elements.

SDP 90-1042 Initiating Document: CSCO 30 Plant Date: N/A Scheduled Completion: 06/28/92

Description: Modify program AUX to run on the new simulator.

SDP 90-1043 Initiating Document: DCP 88-0124/002/00 Plant Date: 05/18/90 Scheduled Completion: 09/15/91

Description: TBCC Auto temp regulator valve to MFP heat exchanger.

SDP 90-1044 Initiating Document: DCP 88-0174/001/00 Plant Date: 03/13/90 Scheduled Completion: 03/13/92

Description: 1N27 - Add temperature control heating element to MFP lube oil reservoir.

SDP 90-1045 Initiating Document: SCR 1-89-1508 Plant Date: 02/05/90 Scheduled Completion: 10/04/91

Description: 1P45 - Raise strainer differential pressure for backwash to 3.75 PSID incr. MPL 1P45N0235

SDP 90-1046 Initiating Document: SCR 1-89-1037 Plant Date: 11/13/89 Scheduled Completion: 10/04/91

Description: 1P45 - Revise MPL 1P45N0251B setpoint.

SDP 90-1047 Initiating Document: SCR 1-89-1045 Plant Date: 01/08/90 Scheduled Completion: 10/04/91

Description: 1P51 - Lower service air compressor autostart setpoint to 107 PSIG dec. MPL 1P51N0185

SDP 90-1048 Initiating Document: SCR 1-89-1046 Plant Date: 01/05/90 Scheduled Completion: 10/04/91

Description: 1P51 - Raise service air receiver tank low pressure alarm to 112 PSIG dec. MPL 1P51N0070

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 90-1049 Initiating Document: SCR 1-89-1047 Plant Date: 01/29/90 Scheduled Completion: 10/04/91

Description: 1P51 - Revise MPL 1P51N0195 setpoint to reflect reset value.

SDP 90-1050 Initiating Document: SCR 0-89-1046 Plant Date: 12/05/89 Scheduled Completion: 10/04/91

Description: 0P43 - Revise MPL 0P43N0351A setpoint and reset value.

SDP 90-1051 Initiating Document: SCR 0-89-1047 Plant Date: 12/13/89 Scheduled Completion: 10/04/91

Description: 0P43 - Revise MPL 0P43N0351B setpoint and reset value.

SDP 90-1052 Initiating Document: SCR 0-89-1048 Plant Date: 02/07/90 Scheduled Completion: 10/04/91

Description: 0P43 - Revise MPL 0P43N0351C setpoint and re[illegible] value.

SDP 90-1053 Initiating Document: SCR 1-89-0070 Plant Date: 02/06/90 Scheduled Completion: 10/04/91

Description: 1D17 - Revise MPL 1D17K0785 setpoints

SDP 90-1054 Initiating Document: SCR 2-89-1019 Plant Date: 01/29/90 Scheduled Completion: 10/04/91

Description: 2P51 - Lower service air compressor autostart setpoint to 107 PSIG dec. MPL 2P51N0185

SDP 90-1055 Initiating Document: SCR 2-89-1021 Plant Date: 01/29/90 Scheduled Completion: 10/04/91

Description: 2P51 - Revise MPL 2P51N0195 setpoint to incorporate reset information.

SDP 90-1056 Initiating Document: SCR 2-89-1022 Plant Date: 01/29/90 Scheduled Completion: 10/04/91

Description: 2P52 - Revise MPL 2P52N0195 setpoint to incorporate reset information.

SDP 90-1057 Initiating Document: SCR 1-90-1014 Plant Date: 02/06/90 Scheduled Completion: 10/04/91

Description: 1E31 - Raise RCIC Equipment Room isolation differential temperature setpoint to 69.4 DEG F. MPL 1E31N0603A

SDP 90-1058 Initiating Document: SCR 1-90-1015 Plant Date: 02/06/90 Scheduled Completion: 10/04/91

Description: 1E31 - Raise RCIC Equipment Room isolation differential temperature setpoint to 69.4 DEG F. MPL 1E31N0603B

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 90-1059 Initiating Document: SCR 1-89-1030 Plant Date: 01/16/90 Scheduled Completion: 10/04/91

Description: 1M41 - Raise MPL 1M41N0280 setpoint to 40 DEG F.

SDP 90-1060 Initiating Document: SCR 1-89-1031 Plant Date: 01/16/90 Scheduled Completion: 10/04/91

Description: 1M41 - Raise MPL 1M41N0281 setpoint to 40 DEG F.

SDP 90-1061 Initiating Document: SCR 1-89-1032 Plant Date: 01/16/90 Scheduled Completion: 10/04/91

Description: 1M41 - Raise MPL 1M41N0282 setpoint to 40 DEG F.

SDP 90-1062 Initiating Document: SCR 1-89-1033 Plant Date: 01/16/90 Scheduled Completion: 10/04/91

Description: 1M41 - Raise MPL 1M41N0283 setpoint to 40 DEG F.

SDP 90-1063 Initiating Document: SCR 1-89-1458 Plant Date: 11/07/89 Scheduled Completion: 10/04/91

Description: 1P42 - Add setpoint for MPL 1P42N0044A.

SDP 90-1064 Initiating Document: SCR 1-89-1459 Plant Date: 11/22/89 Scheduled Completion: 10/04/91

Description: 1P42 - Add setpoint for MPL 1P42N0044B.

SDP 90-1065 Initiating Document: SCR 1-89-1506 Plant Date: 01/22/90 Scheduled Completion: 10/04/91

Description: 1P45 - Raise strainer differential pressure for backwash to 3.75 PSID inc. MPL 1P45N0220A.

SDP 90-1066 Initiating Document: SCR 1-89-1507 Plant Date: 02/23/90 Scheduled Completion: 10/04/91

Description: 1P45 - Raise strainer differential pressure for backwash to 3.75 PSID inc. MPL 1P45N0220B.

SDP 90-1067 Initiating Document: SCR 0-89-0058 Plant Date: 11/28/89 Scheduled Completion: 10/04/91

Description: MPL 0G41N0368A - Raise low level annunciator setpoint to 608'45" which corresponds to a tank level of approximately 90 inches.

SDP 90-1068 Initiating Document: SCR 0-89-0059 Plant Date: 11/28/89 Scheduled Completion: 10/04/91

Description: MPL 0G41N0368B - Raise low level annunciator setpoint to 608'45" which corresnonds to a tank level of approximately 90 inches.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 90-1069 Initiating Document: SCR 1-88-1508 Plant Date: 12/05/89 Scheduled Completion: 10/04/91

Description: MPL 1R23Q0614R - Change the setpoints of bus undervoltage alarm relay NGV13A(27-1) located in EF1D01 as follows: dropout: 96.0V, +4.0V/-1.0V; pickup: 109.0V max.

SDP 90-1070 Initiating Document: SCR 1-88-1566 Plant Date: 12/05/89 Scheduled Completion: 10/04/91

Description: MPL 1E22Q0001 - Change the setpoints of relay 1TE27H(27G) located in 1E22P0001 as follows: pickup: 95.5V, +0.5V/-5.0V; dropout: 86.0V minimum.

SDP 90-1071 Initiating Document: SCR 2-89-0004 Plant Date: 12/05/89 Scheduled Completion: 10/04/91

Description: MPL 2R11Q7001C - Correct setpoint of time delay drop out relay 27-1 (GE type CR2820B120AA4, 480V) to 4.0 +/- 0.2 sec.

SDP 90-1072 Initiating Document: SCR 2-89-0002 Plant Date: 11/07/89 Scheduled Completion: 10/04/91

Description: MPL 2R11Q7001A - Correct setpoint of time delay drop out relay 27-1 (GE type CR2820B120AA4, 480V) to 4.0 +/- 0.2 sec.

SDP 90-1073 Initiating Document: SCR 2-89-0003 Plant Date: 10/16/90 Scheduled Completion: 10/04/91

Description: MPL 2R11Q7001B - Correct setpoint of time delay drop out relay 27-1 (GE type CR2820B120AA4, 480V) to 4.0 +/- 0.2 sec.

SDP 90-1074 Initiating Document: SCR 2-89-0005 Plant Date: 11/07/89 Scheduled Completion: 10/04/91

Description: MPL 2R11Q7002A - Correct setpoint of time delay drop out relay 27-2 (GE type CR2820B120AA4, 480V) to 4.0 +/- 0.2 sec.

SDP 90-1075 Initiating Document: SCR 2-89-0007 Plant Date: 11/07/89 Scheduled Completion: 10/04/91

Description: MPL 2R11Q7002C - Correct setpoint of time delay drop out relay 27-2 (GE type CR2820B120AA4, 480V) to 4.0 +/- 0.2 sec.

SDP 90-1076 Initiating Document: SCR 1-88-1565 Plant Date: 11/13/89 Scheduled Completion: 10/04/91

Description: MPL 1R22Q0804 - Change the setpoints of relay NGV22(59D) located in panel EH1301 as follows: pickup: 95.5V, +0.5V/-5.0V; dropout: 86.0V minimum.

SDP 90-1077 Initiating Document: SCR 1-88-1579 Plant Date: 02/17/90 Scheduled Completion: 10/04/91

Description: MPL 1R42Q0601R - Change the setpoints of relay 12NGV18A2A(27DC) located in ED-1-A as follows: dropout: 113.0V, +/-0.5V; pickup: 130.0V maximum.

SDP 90-1078 Initiating Document: SCR 1-89-1438 Plant Date: 11/07/89 Scheduled Completion: 10/04/91

Description: MPL 1R45N0190B - Change the low, low low and alarm setpoints and reset values for the fuel oil storage tank level transmitter per SCR 1-89-1438.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 90-1079 Initiating Document: SCR 2-89-0006 Plant Date: 11/08/89 Scheduled Completion: 10/04/91

Description: MPL 2R11Q7002B - Correct setpoint of time delay drop out relay 27-2 (GE type CR2820B120AA4, 480V) to 4.0 +/- 0.2 sec.

SDP 90-1080 Initiating Document: SOI-B13/000/01 Plant Date: 09/28/90 Scheduled Completion: 09/15/91

Description: Fabricate and install on 1H13P0883 Operator Aid nameplates for Suppression Pool Temperature Recorders 1D23R0090A & B per TC-1 of SOI-B13 Rev. 0 (Attachment 2).

SDP 90-1081 Initiating Document: DCP 89-0003/001/00 Plant Date: 12/12/90 Scheduled Completion: 09/15/91

Description: Revise B33 Reactor Recirc Pump breakers 3A and 4A control power supply to bus D-1-B (was powered by D-1-A).

SDP 90-1082 Initiating Document: DCP 90-0209/000/00 Plant Date: 11/01/90 Scheduled Completion: 09/15/91

Description: Add to, correct, or relocate various nameplates on Control Room Panels.

SDP 90-1083 Initiating Document: DCN 3317 Plant Date: 10/30/90 Scheduled Completion: 09/15/91

Description: Revise labels on various control room panels. Includes documentation to previous nameplate/annunciator/mimic corrections.

SDP 90-1084 Initiating Document: DCP 90-0086/000/00 Plant Date: 11/15/90 Scheduled Completion: 12/06/91

Description: P42/P45 ESW rotation of 8 spectacle flanges on ECCW system. Affects P42/P45 System models.

SDP 90-1085 Initiating Document: DCP 88-0163/000/00 Plant Date: 11/15/90 Scheduled Completion: 09/15/91

Description: E12 RHR - Add sliding link terminal blocks to eliminate lifting leads during SVI. Affects 1H13-PEI.

SDP 90-1086 Initiating Document: SCR 1-90-1221 Plant Date: 09/25/90 Scheduled Completion: 12/06/91

Description: Decrease MPL 1C34K0636 setpoint (which annunciates 1H13P0680-7-D1) to 1040 psig increasing; LAIZ +/- 4.5 psig; reset >1029.5 psig decreasing.

SDP 90-1087 Initiating Document: SCR 1-90-1138 Plant Date: 09/12/90 Scheduled Completion: 12/06/91

Description: MPL 1S11Q0023 - Add setpoint for Relay SPR (63) located in panel 110-PY-B. Relay setting: Standard (Similar to 1R11Q0004, 1R11Q0014, 1R11Q0024, 1S11Q0015A, B,C)

SDP 90-1088 Initiating Document: DCP 89-0189/001/00 Plant Date: 11/26/90 Scheduled Completion: 12/06/91

Description: M42 Turbine Power Complex vent system - DCP Rev 1 accepts flow of register RPC515 at 290 cfm as balanced.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 90-1089 — Initiating Document: DCN 03119 — Plant Date: 05/24/90 — Scheduled Completion: 09/15/91

Description: Correct label on switch plate R71-S1 (Control Room Lighting, Master Control Switch) on 1H13P0895 to agree with Drawing B-208-219-01 and field conditions.

SDP 90-1090 — Initiating Document: SCR 1-88-1524 — Plant Date: 08/27/90 — Scheduled Completion: 12/19/91

Description: Modify dropout/pickup of MPL 1R22Q0815 relay.

SDP 90-1091 — Initiating Document: SCR 1-89-0007-T — Plant Date: 07/05/90 — Scheduled Completion: 12/19/91

Description: Change NCC Pump to RWCU B lo-flow alarm to 17 gpm (was 20 gpm). Ref. MPL 1P43N0193B

SDP 90-1092 — Initiating Document: DCP 87-0002/001/00 — Plant Date: 12/06/90 — Scheduled Completion: 12/19/91

Description: Holes cut in DW floor drain sump weir box change fill rates with respect to level instrumentation. MPL G61/1E31

SDP 90-1093 — Initiating Document: DCP 90-0072/000/00 — Plant Date: 12/05/90 — Scheduled Completion: 09/15/91

Description: Correct deficiencies in 1H13 panels. (Label changes / corrections)

SDP 90-1094 — Initiating Document: DCN 02840 — Plant Date: 02/15/90 — Scheduled Completion: 12/18/91

Description: Logic to low condenser pressure runbacks. Contact used to close when BPV#1 left full closed position; now contact closes when BPV#1 reaches full open.

SDP 90-1097 — Initiating Document: DCN 3015 — Plant Date: 03/29/90 — Scheduled Completion: 09/15/91

Description: Correct MPL and recorder pen designations to D17 radiation monitors on 1H13P0600.

SDP 90-1098 — Initiating Document: DCN 3134 — Plant Date: 07/11/90 — Scheduled Completion: 12/17/91

Description: Correct operating points for P43.

SDP 91-1001 — Initiating Document: DCP 89-0157/000/0A — Plant Date: 12/23/90 — Scheduled Completion: 01/15/92

Description: Feedwater Seal Injection Pumps 1N27C0005A & B changed from 100 HP to 125 HP motors; impeller increased from 12 1/2" to 13".

SDP 91-1002 — Initiating Document: DCN 03343 Rev. 0 — Plant Date: 11/09/90 — Scheduled Completion: 12/21/91

Description: Optical Isolator on B-208-222 Sh. 103 is given the identity "1:12A-AT2."

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 91-1003 | Initiating Document: DCN 03228 Rev. 0 | Plant Date: 09/10/90 | Scheduled Completion: 12/21/91

Description: MPL swap on B-208-015 Sh. 14 & 15. Per TSR, change in FDS & Data Listings only. Affects S3 variables, setpoints, I/O Map and Instructor Station (I/O override).

SDP 91-1004 | Initiating Document: DCN 03220 Rev. 0 | Plant Date: 09/11/90 | Scheduled Completion: 12/21/91

Description: DG load/time sequence changes incorporated into plant reference drawings.

SDP 91-1005 | Initiating Document: DCP 85-0295/001/01 | Plant Date: 11/27/90 | Scheduled Completion: 12/21/91

Description: Electrical and Instrumentation for Intermediate Building Sub-Exhaust including annunciators and Process Computer points.

SDP 91-1006 | Initiating Document: DCP 89-0066/000/00 | Plant Date: 01/04/91 | Scheduled Completion: 01/15/92

Description: Revision of 1N25 Level Alarm Devices

SDP 91-1007 | Initiating Document: DCP 88-0377/000/00 | Plant Date: 01/21/91 | Scheduled Completion: 01/31/92

Description: 1C85 Bourdon tube type pressure transmitters replaced with Rosemount type 1151 electronic pressure transmitters. Affected MPLs 1C85N0001A/B. Response of loop changes

SDP 91-1008 | Initiating Document: DCP 90-0225/000/00 | Plant Date: 12/11/90 | Scheduled Completion: 12/21/91

Description: Increase pump impeller size for 1P45C0001B.

SDP 91-1009 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 09/15/91

Description: Addition of non-functional relays and wiring harnesses in Panel H13-PEI. Added electrical components in panel will present a more realistic appearance to operators.

SDP 91-1011 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 06/28/92

Description: Restore the ERIS setpoints for pump run status to the plant values.

SDP 91-1012 | Initiating Document: DCN 03413 | Plant Date: 01/10/91 | Scheduled Completion: 01/31/92

Description: Revision to MPL of optical isolator on B-208-222-00414, may affect MF AN01. Previously the drawing showed isolator 1E51-AT5 as an input to annunciator P601-21A-C3.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 91-1013 | Initiating Document: DCN 03192 | Plant Date: 01/02/91 | Scheduled Completion: 02/04/92

Description: Solenoid valves 0M25F0250A/B taken off of S809-354; J930 header. These are supplied through H51P0152 & H51P0153 (J902 & J904 respectively).

SDP 91-1017 | Initiating Document: SCR 1-90-1637 | Plant Date: 01/09/91 | Scheduled Completion: 02/05/92

Description: Change of setpoints for 1C91N0652A/B/C/D. SCR's 1-90-1637 thru 1-90-1640 are covered by this TSR and SDP.

SDP 91-1018 | Initiating Document: SCR 1-90-1154 | Plant Date: 01/09/91 | Scheduled Completion: 02/05/92

Description: SCR changes setpoints for 1E31N0680A/B/E/F. At the time of review, only 1E31N0680F required revision per SCR 1-90-1154.

SDP 91-1019 | Initiating Document: SCR 1-90-1011 | Plant Date: 07/05/90 | Scheduled Completion: 02/05/92

Description: Change to 1D19K0300. This SDP also covers TSR 10781, SCR 1-90-1012, for changes to 1D19K0400.

SDP 91-1020 | Initiating Document: SCR 1-89-1270 | Plant Date: 05/17/90 | Scheduled Completion: 02/05/92

Description: Revise Division 1 Diesel Generator HI/LO Lube Oil Temperature setpoints.

SDP 91-1021 | Initiating Document: SCR 1-89-1008 | Plant Date: 06/07/90 | Scheduled Completion: 02/05/92

Description: Added setpoint for HB ventilation heaters - LO Temp Switch MPL 1M41N0284. Switch added by DCP 88-0293. (Also check setpoints for 1M41N0280, N0281, N0282, & N0283.)

SDP 91-1022 | Initiating Document: SCR 1-90-1008 | Plant Date: 05/14/90 | Scheduled Completion: 02/12/92

Description: Incorporate setpoint 1-90-1008 for 1P12N0010A level switch. This SDP will also incorporate TSR 10890, SCR 1-90-1009 for 1P[illegible]N[illegible]10B level switch.

SDP 91-1025 | Initiating Document: Revision to PEIs | Plant Date: N/A | Scheduled Completion: 06/28/92

Description: Revision of PEIs adds additional operator actions to perform system logic bypass operations (overrides).

SDP 91-1027 | Initiating Document: SCR 1-88-1482 | Plant Date: 12/04/90 | Scheduled Completion: 03/07/92

Description: New setpoint of 1E31N0351A: 135 Deg.F incr. Same for 1E31N0351B (TSR 10976, SCR 1-88-1483); 1E31N0351C (TSR 10979, SCR 1-88-1484); 1E31N0351D (TSR 109[illegible]9, SCR 1-88-1485)

SDP 91-1028 | Initiating Document: SCR 1-90-1013 | Plant Date: 05/09/90 | Scheduled Completion: 03/08/92

Description: Setpoints for summer and winter flow alarms for 1D19K0500.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 91-1029 | Initiating Document: SCR 1-90-1252 | Plant Date: 01/11/91 | Scheduled Completion: 03/07/92

Description: Change setpoint of 1N23N0041 from 35 to 39 psid. Also include SCR 1-90-1253 (TSR 10973) to change setpoint of 1N23N0042 from 30 to 34 psid.

SDP 91-1030 | Initiating Document: SCR 1-90-1632 | Plant Date: 02/11/91 | Scheduled Completion: 03/07/92

Description: Revise setpoint of 1P45N0251B to 7000 gpm.

SDP 91-1031 | Initiating Document: SCR 1-90-1708 | Plant Date: 12/18/90 | Scheduled Completion: 03/07/92

Description: Revise HI/LO alarm setpoint of 1R45N0080.

SDP 91-1032 | Initiating Document: SCR 1-90-1808 | Plant Date: 12/21/90 | Scheduled Completion: 03/07/92

Description: Revise setpoint of 1G61N0025.

SDP 91-1033 | Initiating Document: SCR 1-90-1054 | Plant Date: 12/18/90 | Scheduled Completion: 03/10/92

Description: Modify setpoint for 1G41K0026. This SDP also captures SCR 1-89-1494 for 1G41N0111 (TSR 0011030).

SDP 91-1034 | Initiating Document: SCR 1-90-1709 | Plant Date: 12/18/90 | Scheduled Completion: 03/10/92

Description: Revise setpoints for 1R45N0090 Secondary Fuel Oil Pump. Also; this SDP captures SCR 1-90-1710 for 1R45N0100 (TSR 11022) setpoints for Primary Transfer Pump.

SDP 91-1035 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 09/15/91

Description: Operations request to correct/add three labels for N41 relays on 1H13P0807.

SDP 91-1036 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 09/15/91

Description: Operations request to correct change to nameplate for recorder 1N31R0005 on 1H13P0823.

SDP 91-1037 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 09/15/91

Description: Operations Request added two supplementary operators aides (labels) on 1H13P0632 at recorders 1E31R0608 and 1E31R0611.

SDP 91-1038 | Initiating Document: N/A | Plant Date: N/A | Scheduled Completion: 09/15/91

Description: Three red lamicoid tags with white lettering installed on 1H13P0680 per Operations request.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 91-1040 Initiating Document: DCN 03424 Plant Date: 01/31/91 Scheduled Completion: 09/15/91

Description: Instrument air supply for SCV's OM25F0022A & B are corrected to J-Headers J944 & J943, respectively. These valves supply dampers OM25F0020A & B, respectively.

SDP 91-1041 Initiating Document: DCN 03291 Plant Date: 02/15/91 Scheduled Completion: 03/18/92

Description: Lifted lead #89-00277, documented on DCNs 3208 & 3291, disables clearwell pump trip, OP20C0001A, B, C, on lo-lo basin level as sensed by level switch OP20N0044.

SDP 91-1042 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 09/15/91

Description: Fabricate and attach "dynomarker" tape to identify the location of IRM's and SRM's on the full core display on Unit 2 (1H13P0680).

SDP 91-1043 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 09/15/91

Description: Fabricate and install fuse clips on the 76 fuses for the SRV's and the 8 fuses for the RPS solenoids located in the rear of Unit 21 (1H13-PEI).

SDP 91-1044 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 06/28/92

Description: Add LEDs for "first hit" indication for Reactor Recirculation Flow Control analog lockups (trips) inside rear of panel 1H13P0614.

SDP 91-1045 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 06/28/92

Description: Add 2 red LED's and 2 push buttons to simulate 1B33K0657A&B inside H13P0634. The devices are mounted on cards marked R/C/L in Racks 1 and 2 of Nest 5.

SDP 91-1046 Initiating Document: N/A Plant Date: N/A Scheduled Completion: 06/28/92

Description: Add 4 Flux Estimator Status LEDs (MAINT/FAIL/APRM/EST) in the back of panel 1H13P0634 and make software changes as necessary to drive the lights (4 I/O points).

SDP 91-1047 Initiating Document: SCR 1-91-0007 Plant Date: 01/31/91 Scheduled Completion: 09/15/91

Description: Turbine 1st stage pressure switches set to 212 psig as input for RPS (1C71N0652C). This SDP also covers TSR 11344 for similar change to 1C71N0652D.

SDP 91-1048 Initiating Document: SCR 1-90-1842 Plant Date: 01/15/91 Scheduled Completion: 09/15/91

Description: Pressure setpoint decreased from 126 psig to 114 psig for pressure switch 1N43N0110A - Software only.

SDP 91-1049 Initiating Document: SCR 0-91-1010 Plant Date: 02/19/91 Scheduled Completion: 04/08/92

Description: Revise ALERT setpoint of 0D21K0322 to 14 MR/Hr from 10 MR/Hr. Similar change to 0D21K0332 is included in this SDP under TSR 11288; SCR 0-91-1011.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 91-1050 — Initiating Document: SCR 1-90-1235 — Plant Date: 02/26/91 — Scheduled Completion: 09/15/91

Description: Revise setpoint for 1R42Q0001S time delay to 5 sec. At the time of TSR, S3 Setpoint list gave this value as 2 sec.

SDP 91-1052 — Initiating Document: N/A — Plant Date: N/A — Scheduled Completion: 09/15/91

Description: Obtain switch and knob for unit 02A10 (knob to the left of PPC crt located on 1H13P0680). Install switch and knob on new simulator unit 02A10.

SDP 91-1053 — Initiating Document: N/A — Plant Date: N/A — Scheduled Completion: [illegible]/15/91

Description: Install lead seals and safety wire on simulated and photo mockup relay glass covers on 1H13P0807, 1H13P0808, 1H13P0809, and 1H13P0810. 42 affected relays.

SDP 91-1056 — Initiating Document: SCR 0-90-1035 — Plant Date: 02/19/91 — Scheduled Completion: 04/08/92

Description: Revise setpoint for 0P45N0255B to 565' 7 1/8" (565.59') Elev. at time of TSR; S3 setpoint list gave this value to be 565.34' Elev.

SDP 91-1061 — Initiating Document: N/A — Plant Date: N/A — Scheduled Completion: 06/28/92

Description: Install miscellaneous furniture and equipment in the Simulator Room 123 after installation of the new Simulator. Coordinate with Control Room enhancement activities.

SDP 91-1063 — Initiating Document: SCR 1-89-0010 — Plant Date: 03/07/91 — Scheduled Completion: 09/15/91

Description: Revise (Add) Setpoint for 1C85N0702, emergency high level on EHC fluid reservoir. Switch is an input to annunciator P680-7A-B1 "Steam Bypass HPU Trouble".

SDP 91-1064 — Initiating Document: SCR 1-90-1236 — Plant Date: 03/11/91 — Scheduled Completion: 09/15/91

Description: Change setpoint of 1R42Q0101S to 5 seconds +/- 1 sec. D1B undervoltage time delay relay 27 DCX.

SDP 91-1065 — Initiating Document: SCR 1-90-1234 — Plant Date: 03/15/91 — Scheduled Completion: 09/15/91

Description: Change setpoint of 1R42Q0801S to 5 seconds +/- 1 sec. ED1B undervoltage time delay relay 27 DCX.

SDP 91-1066 — Initiating Document: SCR 1-88-1563 — Plant Date: 04/10/91 — Scheduled Completion: 06/28/92

Description: Pickup and dropout voltages updated for 1R22Q0605

SDP 91-1067 — Initiating Document: SCR 0-91-1020 — Plant Date: 04/04/91 — Scheduled Completion: 06/28/92

Description: Revise alert setpoint of 0D21K0302 to 15 MR/HR (Hi S.P. is 20 MR/HR) Setpoints for 0D21K0332 on TSR 0011702: SCR 0-91-1021 are also included.

Schedule for Correcting Discrepancies Tracked by SDP Number
Section B - To Complete Before June 28, 1992

SDP 91-1068

Initiating Document: SCR 1-90-1237 Plant Date: 05/10/91 Scheduled Completion: 06/28/92

Description: Setpoint for 1R42006015 changed to 5 sec. +/- 1 sec. Bus ED1A undervoltage relay

SDP 91-1069

Initiating Document: DCP 90-0127 Plant Date: N/A Scheduled Completion: 06/28/92

Description: Permanent deactivation of ASB27/59A relays from 4.16kV H11 and H12 normal and alternate sources.

SDP 91-1070

Initiating Document: DCP 90-0011/000/00 Plant Date: 12/24/90 Scheduled Completion: 06/18/92

Description: Revise Off Gas Loop Seal Level instrumentation by replacing level switches 1N64N0016, 1N64N0035, and 1N64N0040 with MPLs 1N64N0772 through 1N64N0789.

TAB C SIMULATOR TESTS

Part 1. Certification Test to ANSI/ANS 3.5 Section Cross Reference

Part 2. Computer Real Time Test Abstract - T.2.7.1

Part 3. Steady State and Normal Operations Test Abstracts

A. Core Performance Tests - T.4.2

B. Continuous Plant Operation Tests - T.4.3

C. Steady State Performance Tests - T.4.4.1

Part 4. Transient Performance Test Abstracts - T.4.4.2

Part 5. Malfunction Test Abstracts

A. System Level Failures - T.4.5.1

B. Malfunction Scenarios - T.4.5.3

ANSI/ANS 3.5 SECTION: 3.1.1 Normal Plant Evolutions

Test T.4.3, CONTINUOUS PLANT OPERATION

ANSI/ANS 3.5 SECTION: 3.1.1 (01) Normal Plant Evolutions; Plant Startup - Cold to Hot Standby

Test T.4.3.1, COLD SHUTDOWN TO REACTOR CRITICAL

ANSI/ANS 3.5 SECTION: 3.1.1 (02) Normal Plant Evolutions; Nuclear Startup from Hot Standby to Rated Power

Test T.4.3.2, REACTOR CRITICAL TO TURBINE SYNCHRONIZED
Test T.4.3.3, POWER INCREASE TO 100% POWER

ANSI/ANS 3.5 SECTION: 3.1.1 (03) Normal Plant Evolutions; Turbine Generator Startup and Synchronization

Test T.4.3.2, REACTOR CRITICAL TO TURBINE SYNCHRONIZED

ANSI/ANS 3.5 SECTION: 3.1.1 (04) Normal Plant Evolutions; Reactor Trip

Test T.4.4.2.01, MANUAL SCRAM

ANSI/ANS 3.5 SECTION: 3.1.1 (05) Normal Plant Evolutions; Operations at Hot Standby

Test T.4.3.2, REACTOR CRITICAL TO TURBINE SYNCHRONIZED

ANSI/ANS 3.5 SECTION: 3.1.1 (06) Normal Plant Evolutions; Load Changes

Test T.4.3.3, POWER INCREASE TO 100% POWER

ANSI/ANS 3.5 SECTION: 3.1.1 (07) Normal Plant Evolutions; Operations with Less than Full Coolant Flow

Test T.4.5.1.27.TH21, POWER/F[illegible] INSTABILITIES (IEB 88-07 SUPPLEMENT 1)
Test T.4.3.3, POWER INCREASE [illegible] 100% POWER

ANSI/ANS 3.5 SECTION: 3.1.1 (08) Normal Plant Evolutions; Shutdown and Cooldown to Cold Shutdown Conditions

Test T.4.3.4, POWER DECREASE TO TURBINE/GENERATOR UNLOADED
Test T.4.3.5, PLANT COOLDOWN TO COLD SHUTDOWN

ANSI/ANS 3.5 SECTION: 3.1.1 (09) Normal Plant Evolutions; Core Performance

Test T.4.2.1, REACTOR CORE XENON TRANSIENT TEST
Test T.4.2.2, CORE FLUX DISTRIBUTION TEST
Test T.4.2.3, CORE THERMAL POWER VS. RECIRC FLOW TEST
Test T.4.2.4, CORE FLUX RESPONSE TO ROD MOVEMENT
Test T.4.2.5, CORE THERMAL PERFORMANCE TEST
Test T.4.2.6, CORE SUBCRITICAL MULTIPLICATION TEST
Test T.4.2.7, REACTOR CORE LIFE TEST
Test T.4.2.8, SHUTDOWN MARGIN DEMONSTRATION
Test T.4.4.1, STEADY STATE PERFORMANCE

ANSI/ANS 3.5 SECTION: 3.1.1 (10) Normal Plant Evolutions; Operator Conducted Surveillances on SR Equipment

Test T.4.3.3, POWER INCREASE TO 100% POWER

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions

Test T.4.5.1, SYSTEM LEVEL FAILURES
Test T.4.5.3, MALFUNCTION SCENARIOS

ANSI/ANS 3.5 SECTION: 3.1.2 (01)(b) Plant Malfunctions; LOCA inside containment

Test T.4.5.1.27.TH01, RECIRC LOOP RUPTURE (DBA LOCA)

ANSI/ANS 3.5 SECTION: 3.1.2 (01)(b) Plant Malfunctions; LOCA outside containment

Test T.4.5.1.04.CU03, REACTOR WATER CLEANUP SYSTEM PIPE BREAK OUTSIDE CONTAINMENT (STEAM TUNNEL)

ANSI/ANS 3.5 SECTION: 3.1.2 (01)(c) Plant Malfunctions; LOCA Large Break

Test T.4.4.2.08, MAXIMUM SIZE LOCA WITH LOSS OF OFFSITE POWER
Test T.4.5.1.27.TH01, RECIRC LOOP RUPTURE (DBA LOCA)

ANSI/ANS 3.5 SECTION: 3.1.2 (01)(c) Plant Malfunctions; LOCA Small Break

Test T.4.5.3.25, PEI MALFUNCTION SCENARIO #4

ANSI/ANS 3.5 SECTION: 3.1.2 (01)(d) Plant Malfunctions; Failure of Safety and Relief valves

Test T.4.5.3.05, INADVERTENT SAFETY/RELIEF VALVE OPENING
Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1

ANSI/ANS 3.5 SECTION: 3.1.2 (02) Plant Malfunctions; Loss of Instrument Air

Test T.4.5.1.12.IA01, AIR RECEIVER LEAK

ANSI/ANS 3.5 SECTION: 3.1.2 (03)(a) Plant Malfunctions; Loss of Off-Site Power

Test T.4.5.3.21, LOSS OF OFF-SITE POWER

ANSI/ANS 3.5 SECTION: 3.1.2 (03)(b) Plant Malfunctions; Loss of Emergency Power

Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)

ANSI/ANS 3.5 SECTION: 3.1.2 (03)(c) Plant Malfunctions; Loss of Diesel Generators

Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)
Test T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3

ANSI/ANS 3.5 SECTION: 3.1.2 (03)(d) Plant Malfunctions; Loss of Distribution Bus

Test T.4.5.1.06.ED06, LOSS OF 480V BUS
Test T.4.5.1.06.ED17, LOSS OF 125V DC DISTRIBUTION PANEL

ANSI/ANS 3.5 SECTION: 3.1.2 (03)(e) Plant Malfunctions; Loss of AC Instrument Bus

Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1

ANSI/ANS 3.5 SECTION: 3.1.2 (03)(e) Plant Malfunctions; Loss of DC Instrument Bus

Test T.4.5.1.06.ED09, LOSS OF 125V DC BUS

ANSI/ANS 3.5 SECTION: 3.1.2 (04)(a) Plant Malfunctions; Loss of Coolant Flow, dual pump

Test T.4.4.2.04, SIMULTANEOUS TRIP OF ALL RECIRC PUMPS

ANSI/ANS 3.5 SECTION: 3.1.2 (04)(a) Plant Malfunctions; Loss of Coolant Flow, single pump

Test T.4.5.3.16, RECIRCULATION PUMP SHAFT SHEAR

ANSI/ANS 3.5 SECTION: 3.1.2 (05)(a) Plant Malfunctions; Loss of Condenser Vacuum

Test T.4.5.1.15.MC01, CONDENSER AIR INLEAKAGE
Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1
Test T.4.5.3.31, EVALUATION MALFUNCTION SCENARIO #5
Test T.4.5.3.36, EVALUATION MALFUNCTION SCENARIO #10

ANSI/ANS 3.5 SECTION: 3.1.2 (06) Plant Malfunctions; Loss of Service Water

Test T.4.5.1.25.SW02, SERVICE WATER SYSTEM PROCESS PIPING LEAKAGE

ANSI/ANS 3.5 SECTION: 3.1.2 (07) Plant Malfunctions; Loss of Shutdown Cooling

Test T.4.5.1.22.RH02, RESIDUAL HEAT REMOVAL SYSTEM PIPE BREAK
Test T.4.5.3.12, FAILURE OF RESIDUAL HEAT REMOVAL SHUTDOWN COOLING
Test T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14

ANSI/ANS 3.5 SECTION: 3.1.2 (08) Plant Malfunctions; Loss of Component Cooling

Test T.4.5.1.25.SW01, NUCLEAR CLOSED COOLING SYSTEM PROCESS PIPING LEAKAGE

ANSI/ANS 3.5 SECTION: 3.1.2 (09) Plant Malfunctions; Normal Feedwater System Failure

Test T.4.5.3.01, LOSS OF FEEDWATER HEATING (REACTOR RECIRC FLOW CONTROL IN AUTO)
Test T.4.5.3.02, LOSS OF FEEDWATER HEATING (REACTOR RECIRC FLOW CONTROL IN MANUAL)
Test T.4.5.3.03, FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND
Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2

ANSI/ANS 3.5 SECTION: 3.1.2 (10) Plant Malfunctions; Loss of All Feedwater

Test T.4.4.2.02, SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS
Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2

ANSI/ANS 3.5 SECTION: 3.1.2 (11) Plant Malfunctions; Loss of Protective System Channel

Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)

ANSI/ANS 3.5 SECTION: 3.1.2 (12)(a) Plant Malfunctions; Stuck Control Rod

Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3
Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1

ANSI/ANS 3.5 SECTION: 3.1.2 (12)(b) Plant Malfunctions; Uncoupled Control Rod

Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1

ANSI/ANS 3.5 SECTION: 3.1.2 (12)(c) Plant Malfunctions; Drifting Control Rod

Test T.4.5.1.21.RD03, CONTROL ROD DRIFT - IN
Test T.4.5.1.21.RD04, CONTROL ROD DRIFT - OUT

ANSI/ANS 3.5 SECTION: 3.1.2 (12)(d) Plant Malfunctions; Control Rod Drop

Test T.4.5.3.18, CONTROL ROD DROP ACCIDENT

ANSI/ANS 3.5 SECTION: 3.1.2 (13) Plant Malfunctions; Inability to Move Control Rods

Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Test T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8
Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11

ANSI/ANS 3.5 SECTION: 3.1.2 (14) Plant Malfunctions; Fuel Cladding Failure

Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)

ANSI/ANS 3.5 SECTION: 3.1.2 (15) Plant Malfunctions; Turbine Trip

Test T.4.5.1.28.TU01, MAIN SHAFT OIL PUMP DEGRADATION
Test T.4.5.3.10, TURBINE TRIP

ANSI/ANS 3.5 SECTION: 3.1.2 (16) Plant Malfunctions; Generator Trip

Test T.4.5.1.07.EG01, MAIN GENERATOR LOCKOUT RELAY TRIP
Test T.4.5.3.08, GENERATOR LOAD REJECT WITH BYPASS VALVES
Test T.4.5.3.09, GENERATOR LOAD REJECT WITHOUT BYPASS VALVES
Test T.4.5.3.25, PEI MALFUNCTION SCENARIO #4

ANSI/ANS 3.5 SECTION: 3.1.2 (17) Plant Malfunctions; Failure in Reactivity Control System

Test T.4.5.1.23.RP02, INADVERTENT INITIATION OF ALTERNATE ROD INSERTION
Test T.4.5.3.13, RECIRC FLOW CONTROL FAILURE-DECREASING (BOTH FCV'S)
Test T.4.5.3.17, RECIRC FLOW CONTROL FAILURE-INCREASING (BOTH FCV'S)
Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2

ANSI/ANS 3.5 SECTION: 3.1.2 (19) Plant Malfunctions; Reactor Trip

Test T.4.4.2.01, MANUAL SCRAM

ANSI/ANS 3.5 SECTION: 3.1.2 (20)(a) Plant Malfunctions; Main Steam Line Break, inside containment

Test T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12

ANSI/ANS 3.5 SECTION: 3.1.2 (20)(b) Plant Malfunctions; Main Steam Line Break, outside containment

Test T.4.5.1.27.TH27, MAIN STEAM LINE RUPTURE IN STEAM TUNNEL

ANSI/ANS 3.5 SECTION: 3.1.2 (20)(c) Plant Malfunctions; Feedwater Line Break, inside containment

Test T.4.5.1.09.FW02, FEEDWATER SYSTEM PIPE BREAK INSIDE DRYWELL

ANSI/ANS 3.5 SECTION: 3.1.2 (20)(c) Plant Malfunctions; Feedwater Line Break, outside containment

Test T.4.5.1.09.FW03, FEEDWATER SYSTEM PIPE BREAK OUTSIDE CONTAINMENT

ANSI/ANS 3.5 SECTION: 3.1.2 (21) Plant Malfunctions; Nuclear Instrumentation Failures

Test T.4.5.1.17.NM01, SRM DETECTOR (PRE-AMP) FAILURE
Test T.4.5.1.17.NM02, IRM DETECTOR (PRE-AMP) FAILURE
Test T.4.5.1.17.NM03, LPRM DETECTOR FAILURE
Test T.4.5.1.17.NM10, NEUTRON MONITORING DETECTOR DRIVE STUCK
Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2
Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #6

ANSI/ANS 3.5 SECTION: 3.1.2 (22)(a) Plant Malfunctions; Process Instrument System Failures

Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3
Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1
Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2
Test T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3
Test T.4.5.3.30, EVALUATION MALFUNCTION SCENARIO #4
Test T.4.5.3.33, EVALUATION MALFUNCTION SCENARIO #7
Test T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8
Test T.4.5.3.35, EVALUATION MALFUNCTION SCENARIO #9
Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11
Test T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12
Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13
Test T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14
Test T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15
Test T.4.5.3.42, PEI MALFUNCTION SCENARIO #5 (PART 2)

ANSI/ANS 3.5 SECTION: 3.1.2 (22)(b) Plant Malfunctions; Process Alarm System Failure

Test T.4.5.1.02.AN01, ANNUNCIATOR INPUT OPTICAL ISOLATOR FAILURE

ANSI/ANS 3.5 SECTION: 3.1.2 (22)(c) Plant Malfunctions; Process Control System Failures

Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)
Test T.4.5.3.30, EVALUATION MALFUNCTION SCENARIO #4
Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13

ANSI/ANS 3.5 SECTION: 3.1.2 (23) Plant Malfunctions; Passive and Active Component Failure in Plant Systems

Test T.4.5.3.08, GENERATOR LOAD REJECT WITH BYPASS VALVES
Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3
Test T.4.5.3.25, PEI MALFUNCTION SCENARIO #4
Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)
Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1
Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2
Test T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3
Test T.4.5.3.30, EVALUATION MALFUNCTION SCENARIO #4
Test T.4.5.3.31, EVALUATION MALFUNCTION SCENARIO #5
Test T.4.5.3.32, EVALUATION MALFUNCTION SCENARIO #6
Test T.4.5.3.33, EVALUATION MALFUNCTION SCENARIO #7
Test T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8
Test T.4.5.3.35, EVALUATION MALFUNCTION SCENARIO #9
Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11
Test T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12
Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13
Test T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14
Test T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15
Test T.4.5.3.42, PEI MALFUNCTION SCENARIO #5 (PART 2)

ANSI/ANS 3.5 SECTION: 3.1.2 (24) Plant Malfunctions; Failure of the Automatic Reactor Trip System

Test T.4.5.3.20, ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)
Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3
Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)
Test T.4.5.3.36, EVALUATION MALFUNCTION SCENARIO #10
Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11
Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13
Test T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15
Test T.4.5.3.42, PEI MALFUNCTION SCENARIO #5 (PART 2)

ANSI/ANS 3.5 SECTION: 3.1.2 (25) Plant Malfunctions; Reactor Pressure Control System Failure (BWR)

Test T.4.5.3.04, PRESSURE REGULATOR FAILURE-OPEN
Test T.4.5.3.07, PRESSURE REGULATOR FAILURE-CLOSED

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.1.1

Test T.4.5.3.01, LOSS OF FEEDWATER HEATING (REACTOR RECIRC FLOW CONTROL IN AUTO)
Test T.4.5.3.02, LOSS OF FEEDWATER HEATING (REACTOR RECIRC FLOW CONTROL IN MANUAL)

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.1.2

Test T.4.5.3.03, FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.1.3

Test T.4.5.3.04, PRESSURE REGULATOR FAILURE-OPEN

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.1.4

Test T.4.5.3.05, INADVERTENT SAFETY/RELIEF VALVE OPENING

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.1.6

Test T.4.5.3.06, INADVERTENT RESIDUAL HEAT REMOVAL SHUTDOWN COOLING OPERATION

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.1

Test T.4.5.3.07, PRESSURE REGULATOR FAILURE-CLOSED

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.10

Test T.4.5.1.12.IA02, INSTRUMENT AIR LINE LEAK

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.2

Test T.4.5.3.08, GENERATOR LOAD REJECT WITH BYPASS VALVES
Test T.4.5.3.09, GENERATOR LOAD REJECT WITHOUT BYPASS VALVES

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.3

Test T.4.5.3.10, TURBINE TRIP

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.4

Test T.4.4.2.03, SIMULTANEOUS CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
Test T.4.4.2.10, SIMULTANEOUS CLOSURE OF MAIN STEAM ISOLATION VALVES W/SINGLE STUCK OPEN SAFETY/RELIEF VALVE

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.5

Test T.4.5.1.15.MC01, CONDENSER AIR INLEAKAGE

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.6

Test T.4.5.3.08, GENERATOR LOAD REJECT WITH BYPASS VALVES
Test T.4.5.3.11, LOSS OF AC POWER (LOSS OF AUX TRANSFORMER)
Test T.4.5.3.21, LOSS OF OFF-SITE POWER

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.7

Test T.4.4.2.02, SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.8

Test T.4.4.2.09, MAXIMUM SIZE UNISOLABLE MAIN STEAM LINE RUPTURE

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.2.9

Test T.4.5.1.22.RH02, RESIDUAL HEAT REMOVAL SYSTEM PIPE BREAK
Test T.4.5.3.12, FAILURE OF RESIDUAL HEAT REMOVAL SHUTDOWN COOLING

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.3.1

Test T.4.4.2.04, SIMULTANEOUS TRIP OF ALL RECIRC PUMPS
Test T.4.4.2.05, SINGLE RECIRC PUMP TRIP

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.3.2

Test T.4.5.3.13, RECIRC FLOW CONTROL FAILURE-DECREASING (BOTH FCV'S)

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.3.3

Test T.4.5.3.14, RECIRCULATION PUMP SEIZURE

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.3.4

Test T.4.5.3.16, RECIRCULATION PUMP SHAFT SHEAR

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.4.4

Test T.4.5.3.15, ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.4.5

Test T.4.5.3.17, RECIRC FLOW CONTROL FAILURE-INCREASING (BOTH FCV'S)

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.4.9

Test T.4.5.3.18, CONTROL ROD DROP ACCIDENT

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.5.1

Test T.4.5.3.19, INADVERTENT HPCS STARTUP

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.5.3

Test T.4.5.3.03, FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.6.1

Test T.4.5.3.05, INADVERTENT SAFETY/RELIEF VALVE OPENING

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.6.2

Test T.4.5.1.27.TH19, RPV LEVEL INST REFERENCE LEG BREAK
Test T.4.5.1.27.TH20, RPV LEVEL INST VARIABLE LEG BREAK

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.6.4

Test T.4.5.1.27.TH27, MAIN STEAM LINE RUPTURE IN STEAM TUNNEL

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.6.5

Test T.4.4.2.08, MAXIMUM SIZE LOCA W/LOSS OFFSITE POWER

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.6.6

Test T.4.5.1.09.FW02, FEEDWATER SYSTEM PIPE BREAK INSIDE DRYWELL
Test T.4.5.1.09.FW03, FEEDWATER SYSTEM PIPE BREAK OUTSIDE CONTAINMENT

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.7.1

Test T.4.5.1.16.MS11, STEAM SEAL HEADER PRESSURE REGULATOR FAILURE
Test T.4.5.1.18.OG03, OFF GAS SYSTEM LEAK UPSTREAM ADSORBERS
Test T.4.5.1.18.OG04, OFF GAS SYSTEM LEAK DOWNSTREAM ADSORBERS

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions, USAR Accident 15.8

Test T.4.5.3.20, ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

ANSI/ANS 3.5 SECTION: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Test T.4.5.1.27.TH21, POWER/FLOW INSTABILITIES (IEB 88-07 SUPPLEMENT 1)
Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3
Test T.4.5.3.25, PEI MALFUNCTION SCENARIO #4
Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)
Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1
Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2
Test T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3
Test T.4.5.3.30, EVALUATION MALFUNCTION SCENARIO #4
Test T.4.5.3.31, EVALUATION MALFUNCTION SCENARIO #5
Test T.4.5.3.32, EVALUATION MALFUNCTION SCENARIO #6
Test T.4.5.3.33, EVALUATION MALFUNCTION SCENARIO #7
Test T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8
Test T.4.5.3.35, EVALUATION MALFUNCTION SCENARIO #9
Test T.4.5.3.36, EVALUATION MALFUNCTION SCENARIO #10
Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11
Test T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12
Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13
Test T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14
Test T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15
Test T.4.5.3.42, PEI MALFUNCTION SCENARIO #5 (PART 2)

ANSI/ANS 3.5 SECTION: Appendix A 3.1 Computer Real Time Test

Test T.2.7.1, SPARE TIME VERIFICATION

ANSI/ANS 3.5 SECTION: Appendix A 3.2 Normal Operations

Test T.4.3, CONTINUOUS PLANT OPERATION

ANSI/ANS 3.5 SECTION: Appendix A 3.2 Steady State Performance

Test T.4.4.1, STEADY STATE PERFORMANCE
Test T.4.4.1.1, 25% POWER HEAT BALANCE
Test T.4.4.1.2, 50% POWER HEAT BALANCE
Test T.4.4.1.3, 75% POWER HEAT BALANCE
Test T.4.4.1.4, 100% POWER HEAT BALANCE
Test T.4.4.1.5, 100% POWER STABILITY TEST

ANSI/ANS 3.5 SECTION: Appendix A 3.3 Transient Tests

Test T.4.4.2, TRANSIENT PERFORMANCE
Test T.4.4.2.01, MANUAL SCRAM
Test T.4.4.2.02, SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS
Test T.4.4.2.03, SIMULTANEOUS CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
Test T.4.4.2.04, SIMULTANEOUS TRIP OF ALL RECIRC PUMPS
Test T.4.4.2.05, SINGLE RECIRC PUMP TRIP
Test T.4.4.2.06, MAIN TURBINE TRIP W/O REACTOR SCRAM
Test T.4.4.2.07, MAXIMUM RATE POWER RAMP (100% - 75% - 100%)
Test T.4.4.2.08, MAXIMUM SIZE LOCA W/LOSS OF OFFSITE POWER

Test T.4.4.2.09, MAXIMUM SIZE UNISOLABLE MAIN STEAM LINE RUPTURE
Test T.4.4.2.10, SIMULTANEOUS CLOSURE OF MAIN STEAM ISOLATION VALVES W/SINGLE STUCK OPEN SAFETY/RELIEF VALVE

ANSI/ANS 3.5 SECTION: Appendix A 3.4 Malfunction Tests

Test T.4.5, MALFUNCTION AND COMPONENT FAILURE TESTS

ANSI/ANS 3.5 SECTION: Appendix B Simulator Operability Tests

Test T.4.4, BWR SIMULATOR OPERABILITY TESTS

ANSI/ANS 3.5 SECTION: Appendix B 1.1 Steady State Performance

Test T.4.4.1, STEADY STATE PERFORMANCE
Test T.4.4.1.1, 25% POWER HEAT BALANCE
Test T.4.4.1.2, 50% POWER HEAT BALANCE
Test T.4.4.1.3, 75% POWER HEAT BALANCE
Test T.4.4.1.4, 100% POWER HEAT BALANCE
Test T.4.4.1.5, 100% POWER STABILITY TEST

ANSI/ANS 3.5 SECTION: Appendix B 1.2 Transient Performance

Test T.4.4.2, TRANSIENT PERFORMANCE

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (01) Transient Performance

Test T.4.4.2.01, MANUAL SCRAM

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (02) Transient Performance

Test T.4.4.2.02, SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (03) Transient Performance

Test T.4.4.2.03, SIMULTANEOUS CLOSURE OF ALL MAIN STEAM ISOLATION VALVES

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (04) Transient Performance

Test T.4.4.2.04, SIMULTANEOUS TRIP OF ALL RECIRC PUMPS

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (05) Transient Performance

Test T.4.4.2.05, SINGLE RECIRC PUMP TRIP

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (06) Transient Performance

Test T.4.4.2.06, MAIN TURBINE TRIP W/O REACTOR SCRAM

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (07) Transient Performance

Test T.4.4.2.07, MAXIMUM RATE POWER RAMP (100% - 75% - 100%)

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (08) Transient Performance

Test T.4.4.2.08, MAXIMUM SIZE LOCA W/LOSS OF OFFSITE POWER

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (09) Transient Performance

Test T.4.4.2.09, MAXIMUM SIZE UNISOLABLE MAIN STEAM LINE RUPTURE

ANSI/ANS 3.5 SECTION: Appendix B 1.2 (10) Transient Performance

Test T.4.4.2.10, SIMULTANEOUS CLOSURE OF MAIN STEAM ISOLATION VALVES W/SINGLE STUCK OPEN SAFETY/RELIEF VALVE

ATP TEST SECTION T.2.7.1

COMPUTER REAL TIME TEST ABSTRACT

Test T.2.7.1, SPARE TIME VERIFICATION
Revision Number: Later *
ANSI/ANS 3.5 Section: Appendix A 3.1 Computer Real Time Test

Date Tested: Not RUN *

Run Time: Later hours *

Test Description: This test measures the available spare time on each of the main simulation computer nodes while the simulator is running a scenario containing a large number of malfunctions and involves the interaction of as many BOP and Safety systems as possible. Two measurements are taken in each node (CPU): total spare time, and spare time execution per frame. The criteria for each CPU are: a minimum of 40% spare time total, and a minimum of 20% spare time per frame. The test has no effect on the simulation and runs as a background task.

Baseline Data used For Reference

Reference Type: Other
Simulator Specification

Malfunctions Tested: None

Discrepancies: Unknown *

Evaluators: Later *

* This test has not been completed (N/C).

ATP TEST SECTION T.4.2

CORE PERFORMANCE TESTS

Overview of testing performed in this section

This section of testing verifies the nuclear characteristics of the simulated core model are in compliance with established laws of nuclear physics and also matches the parameters of the Perry Cycle 1 core. Separate tests are run to determine the relative reactivities associated with fission product poisons and control rods. Neutron Flux profiles, thermal performance and flux response to core flow (power to flow map) are also checked at power. The shutdown margin verification and sub-critical multiplication tests are performed on the shutdown core.

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.2.1, REACTOR CORE XENON TRANSIENT TEST	2	None
T.4.2.2, CORE FLUX DISTRIBUTION TEST	3	1*
T.4.2.3, CORE THERMAL POWER VS. RECIRC FLOW TEST	4	None
T.4.2.4, CORE FLUX RESPONSE TO ROD MOVEMENT	5	None
T.4.2.5, CORE THERMAL PERFORMANCE TEST	6	1*
T.4.2.6, CORE SUBCRITICAL MULTIPLICATION TEST	7	None
T.4.2.7, REACTOR CORE LIFE TEST	8	None
T.4.2.8, SHUTDOWN MARGIN DEMONSTRATION	9	None

ANSI/ANS 3.5 Reference: *3.1.1 (09) Normal Plant Evolutions; Core Performance*

* This test has not been completed (N/C) or has been performed but has one or more unresolved major Simulator Discrepancy Reports associated with it. Please see the individual test abstract for these discrepancies which constitute exceptions to ANSI/ANS 3.5 section 3.1.1.

Test T.4.2.1, REACTOR CORE XENON TRANSIENT TEST
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

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Date Tested: 06/04/91

Run Time: 8.00 hours

Test Description: This test verifies the nuclear characteristics of the simulated core model with respect to the reactivity effects of the major fission product poison, Xenon.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION

Malfunctions Tested: None

Discrepancies: None

Evaluators: B. Panfil, B. Stetson

Test T.4.2.2, CORE FLUX DISTRIBUTION TEST
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

---

Date Tested: 04/23/91

Run Time: 1.50 hours

Test Description: This test verifies that the simulated core axial flux shape is correct for the BOC and EOC cores.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION

Malfunctions Tested: None

Major Discrepancies:

F-0169 CANNOT PERFORM (PASS) T.4.2.2 AND T.4.2.5 DUE TO MFLCPR GREATER THAN 1. SHOULD BE ABLE TO OPERATE ON PCTLLP UP TO 110% WITHOUT LIMIT VIOLATIONS.

Evaluators: P. Bordley, C. Persson, B. Panfil, R. Kearney, G. Minshall, J. Pierson

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Test T.4.2.3, CORE THERMAL POWER VS. RECIRC FLOW TEST
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

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Date Tested: 06/10/91

Run Time: 3.00 hours

Test Description: This test verifies that the power to flow characteristics of the core model are valid in both forced circulation and natural circulation, and verified using the Perry Unit One Power to Flow Map.

Baseline Data used For Reference

Reference Type: Plant Dat
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION
PDB-A006 POWER TO FLOW MAP
PDB-A010 LOOP FLOW VERSUS FCV POSITION
PDB-A012 TOTAL CORE FLOW VERSUS RECIRCULATION DRIVE FLOW

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, H. DeBoer

Test T.4.2.4, CORE FLUX RESPONSE TO ROD MOVEMENT
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

---

Date Tested: 06/05/91

Run Time: 3.00 hours

Test Description: This test verifies that the movement of shallow, intermediate, and deep control rods, produce reactor power changes of the correct direction and magnitude.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION

Malfunctions Tested: None

Discrepancies: None

Evaluators: B. Panfil, B. Stetson

Test T.4.2.5, CORE THERMAL PERFORMANCE TEST
Revision Number: C2
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

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Date Tested: 04/24/91

Run Time: 5.00 hou.s

Test Description: This test verifies that core thermal limits are within Perry Unit 1 Technical Specification limits at various combinations of reactor power and core flow.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION
Technical Specifications PERRY UNIT 1 TECHNICAL SPECIFICATIONS

Malfunctions Tested: None

Major Discrepancies:

P-1335 DURING CORE TEST TO VERIFY THAT THERMAL LIMITS WERE NOT EXCEEDED DURING OPERATION ALLOWED BY THE POWER/FLOW MAP, THERMAL LIMITS WERE EXCEEDED AT SOME CORE FLOW/RX POWER CONDITIONS. THERMAL LIMITS SHOULD NOT BE EXCEEDED.

Evaluators: C. Persson, B. Panfil, R. Kearney, G. Minshall, J. Pierson

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Test T.4.2.6, CORE SUBCRITICAL MULTIPLICATION TEST
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

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Date Tested: 05/23/91

Run Time: 0.50 hours

Test Description: This test verifies the proper response of the core model neutron population during a reactor startup with power in the source range.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION

Malfunctions Tested: None

Discrepancies: None

Evaluators: C. Persson, B. Panfil, J. Steward, B. Stetson, J. McHugh

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Test T.4.2.7, REACTOR CORE LIFE TEST
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

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Date Tested: 05/23/91

Run Time: 2.50 hours

Test Description: This test verifies that the amount of control rod withdrawal needed to reach criticality, as well as Nuclear Instrumentation response, change appropriately as core age increases.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION

Malfunctions Tested: None

Discrepancies: None

Evaluators: C. Persson, B. Panfil, J. Steward, B. Stetson, J. McHugh

Test T.4.2.8, SHUTDOWN MARGIN DEMONSTRATION
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (09) Normal Plant Evolutions; Core Performance

Date Tested: 05/23/91

Run Time: 0.50 hours

Test Description: This test verifies that the shutdown margin for the simulated BOL and EOL cores comply with Perry Unit 1 Technical Specification limits.

Baseline Data used For Reference

Reference Type: Plant Data
Cycle 1 Benchmark Data GENERATED BY PERRY REACTOR ENGINEERING SECTION
Technical Specifications PERRY UNIT 1 TECHNICAL SPECIFICATIONS

Malfunctions Tested: None

Discrepancies: None

Evaluators: C. Persson, B. Panfil, J. Steward, B. Stetson, J. McHugh

ATP TEST SECTION T.4.3

CONTINUOUS PLANT OPERATION

Overview of testing performed in this section

The purpose of this series of tests is to demonstrate the simulator's ability to perform a complete nuclear plant startup and shutdown starting from cold shutdown conditions (with a minimum of support systems operating) up to 100% rated power, and back to the cold shutdown condition. The tests were written directly from the PNPP Unit 1 Operations Manual, and only operations directed by procedures in use in the Perry Unit 1 Control Room are performed. The tests are written such that the entire startup and shutdown evolution is continuous, with each test proceeding from the end of the prior test. All indicators, computer readouts, logs, alarms, and other forms of man-machine interface available in the Perry Unit 1 Control Room were verified to the criteria of ANSI/ANS 3.5. All operator conducted surveillances and other periodic tests were also tested within this section. These tests were performed per section 3.2(2) of Appendix A of ANSI/ANS 3.5 as required by section 5.4.1 for initial construction performance testing.

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.3.1, COLD SHUTDOWN TO REACTOR CRITICAL	2	9*
T.4.3.2, REACTOR CRITICAL TO TURBINE SYNCHRONIZED	5	14*
T.4.3.3, POWER INCREASE TO 100% POWER	7	9*
T.4.3.4, POWER DECREASE TO TURBINE/GENERATOR UNLOADED	10	None
T.4.3.5, PLANT COOLDOWN TO COLD SHUTDOWN	12	8*

ANSI/ANS 3.5 Reference: *Appendix A 3.2 Normal Operations*

* This test has not been completed (N/C) or has been performed but has one or more unresolved major Simulator Discrepancy Reports associated with it. Please see the individual test abstract for these discrepancies which constitute exceptions to ANSI/ANS 3.5 section 3.1.

Test T.4.3.1, COLD SHUTDOWN TO REACTOR CRITICAL
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (01) Normal Plant Evolutions; Plant Startup - Cold to Hot Standby

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Date Tested: 05/22/91

Run Time: 14.20 hours

Test Description: The purpose of this test is to demonstrate the simulator's ability to conduct a normal plant startup from cold iron conditions using only operator action normal to Perry Unit 1. At the beginning of the test, the reactor is shutdown, as are most supporting systems (IC01). Supporting BOP systems are started per the cold startup lineup (attachment 1 to IOI-1). Safety systems are placed in the Standby Readiness mode. A reactor startup to reactor critical is then performed per Perry Operations Manual Integrated Operating Instruction IOI-1. All instructions used as references for this test are the same ones used by Operators in the Perry Unit 1 Control Room. At the completion of the test, the reactor is critical and ready for Nuclear Heatup.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
- FTI-B02 CONTROL ROD MOVEMENTS
- IOI-1 COLD STARTUP
- SOI-B21 NUCLEAR STM SUPPLY SHUTOFF, AUTO DEPRESSURIZATION, & NSSS
- SOI-B33 REACTOR RECIRCULATION SYSTEM
- SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
- SOI-C11(RCIS) CONTROL ROD & INFORMATION SYSTEM (UNIT 1)
- SOI-C85 STEAM BYPASS & PRESSURE REGULATOR SYSTEM (UNIT 1)
- SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
- SOI-E21 LOW PRESSURE CORE SPRAY SYSTEM (UNIT 1)
- SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
- SOI-E22B DIVISION 3 DIESEL GENERATOR (UNIT 1)
- SOI-E31 LEAK DETECTION SYSTEM (UNIT 1)
- SOI-G33 REACTOR WATER CLEANUP SYSTEM (UNIT 1)
- SOI-G41(FPCC) FUEL POOL COOLING & CLEANUP SYSTEM (UNIT 1)
- SOI-G41(FPFD) FUEL POOL FILTER DEMINERALIZER SYSTEM (UNIT 1)
- SOI-M11 CONTAINMENT VESSEL COOLING SYSTEM (UNIT 1)
- SOI-M13 DRYWELL COOLING SYSTEM (UNIT 1)
- SOI-M14 CONTAINMENT VESSEL & DRYWELL PURGE SYSTEM (UNIT 1)
- SOI-M15 ANNULUS EXHAUST GAS TREATMENT SYSTEM (UNIT 1)
- SOI-M21 CONTROLLED ACCESS & MISC EQUIPMENT AREA (CA&MEA) HVAC SYSTEMS
- SOI-M23/24 MCC, SWGR, & MISC ELECTRICAL EQUIPMENT AREA HVAC SYSTEMS
- SOI-M25/26 CONTROL ROOM HVAC & EMERGENCY RECIRCULATION SYSTEM
- SOI-M28 EMERGENCY CLOSED COOLING PUMP AREA COOLING SYSTEM
- SOI-M32 EMERGENCY SERVICE WATER PUMP HOUSE VENTILATION SYSTEM (UNIT 1)
- SOI-M33 INTERMEDIATE BUILDING VENTILATION SYSTEM

Test T.4.3.1, COLD SHUTDOWN TO REACTOR CRITICAL

SOI-M35 TURBINE BUILDING VENTILATION SYSTEM (UNIT 1)
SOI-M36 OFF-GAS BUILDING EXHAUST SYSTEM (UNIT 1)
SOI-M38 AUXILIARY BUILDING VENTILATION SYSTEM (UNIT 1)
SOI-M39 ECCS PUMP ROOMS COOLING SYSTEM (UNIT 1)
SOI-M40 FUEL HANDLING BUILDING VENTILATION SYSTEM
SOI-M41 HEATER BAY VENTILATION SYSTEM (UNIT 1)
SOI-M42 TURBINE POWER COMPLEX VENTILATION SYSTEM
SOI-M43 DIESEL GENERATOR BUILDING VENTILATION SYSTEM (UNIT 1)
SOI-M47 STEAM TUNNEL COOLING SYSTEM (UNIT 1)
SOI-N11 MAIN & REHEAT STEAM SYSTEM (UNIT 1)
SOI-N21 CONDENSATE SYSTEM (UNIT 1)
SOI-N23 CONDENSATE FILTRATION SYSTEM (UNIT 1)
SOI-N24 CONDENSATE DEMINERALIZER SYSTEM (UNIT 1)
SOI-N27 FEEDWATER SYSTEM (UNIT 1)
SOI-N32/39 MAIN TURBINE & TURNING GEAR SYSTEM (UNIT 1)
SOI-N33 STEAM SEAL SYSTEM (UNIT 1)
SOI-N34 MAIN LUBE OIL SYSTEM (UNIT 1)
SOI-N42 HYDROGEN SEAL OIL SYSTEM (UNIT 1)
SOI-N43 STATOR WATER COOLING SYSTEM (UNIT 1)
SOI-N64(OGVRS) OFF-GAS VAULT REFRIGERATION SYSTEM (UNIT 1)
SOI-N64/62 OFF-GAS/CONDENSER AIR REMOVAL SYSTEM
SOI-N71 CIRCULATING WATER/CONDENSER MECHANICAL CLEANING SYSTEM (UNIT 1)
SOI-P11 CONDENSATE TRANSFER & STORAGE SYSTEM (UNIT 1)
SOI-P12 CONDENSATE SEAL SYSTEM (UNIT 1)
SOI-P40/41 SERVICE WATER & SERVICE WATER SCREEN WASH
SOI-P42 EMERGENCY CLOSED COOLING SYSTEM (UNIT 1)
SOI-P43 NUCLEAR CLOSED COOLING SYSTEM
SOI-P44 TURBINE BUILDING CLOSED COOLING SYSTEM (UNIT 1)
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)
SOI-P46 TURBINE BUILDING CHILLED WATER SYSTEM (UNIT 1)
SOI-P47 CONTROL COMPLEX CHILLED WATER SYSTEM
SOI-P50 CONTAINMENT VESSEL CHILLED WATER SYSTEM (UNIT 1)
SOI-P54 FIRE PROTECTION SYSTEM
SOI-R13 ISOLATION PHASE BUS DUCT COOLING SYSTEM (UNIT 1)
SOI-R43 DIVISION 1 & 2 DIESEL GENERATOR SYSTEM (UNIT 1)
SOI-R44 DIVISION 1 & 2 DIESEL GENERATOR STARTING AIR SYSTEM (UNIT 1)
SOI-R44/E22B DIVISION 3 DIESEL GENERATOR STARTING AIR SYSTEM (UNIT 1)
SOI-R45 DIVISION 1 & 2 DIESEL GENERATOR FUEL OIL SYSTEM (UNIT 1)
SOI-R45/E22B DIVISION 3 DIESEL GENERATOR FUEL OIL SYSTEM (UNIT 1)

Reference Type: Surveillance Procedures
SVI-B33-T1160 JET PUMP OPERABILITY
SVI-C11-T1019 ROD PATTERN CONTROLLER SYS TEST BELOW LOW POWER SETPOINT
SVI-M14-T2003 CONTAINMENT INBOARD & DRYWELL PURGE SUPPLY/EXHAUST ISOLATION DAMPERS OPERABILITY TEST

Test T.4.3.1, COLD SHUTDOWN TO REACTOR CRITICAL

Malfunctions Tested: None

Major Discrepancies:

F-0003	M14 SYSTEM NEVER SHIFTED TO THE DP CONTROLLER.
F-0024	UNABLE TO KEEP CRDH PUMP RUNNING ON INITIAL START, TRIPS ON LOW SUCTION PRESSURE.
F-0032	WITH NO FLOW THROUGH THE OFF GAS SYSTEM, COOLER CONDENSER OUTLET TEMP "B" WENT TO 40 DEG AS SOON AS GLYCOL WAS COOLED DOWN AND "A" SHOWED NO CHANGE.
F-0034	WHEN OFF GAS RECOMBINERS HEATERS WERE STARTED, RECOMB A TEMP SHOWED NO INCREASE OVER 3 HOURS AND B WENT TO NORMAL TEMP PROFILE IN 5 MIN.
F-0035	THE VENTING OF RR PUMP SEALS HAD AN INAPPROPRIATE EFFECT ON CRD AND ON RR SYSTEM (OUTER SEAL LEAKAGE).
F-0039	CRD FLOW/PRESSURE OSCILLATIONS AND INTERMITTENT SUCTION FILTER DP ALARM. HI DP ALARM NOT CLEARING UNTIL FLOW IS <50 GPM.
F-0048	WHEN ESTABLISHING FLOW TO HOT SURGE TANK, OBSERVED FLOW, PRESS INCREASES WERE SEEN ON CRDH SYS W/ NO CRDH PUMP IN OPERATION.
F-0049	WHEN THE MECHANICAL VACUUM PUMPS WERE STARTED, OBSERVED NO CHANGE IN OG VENT FLOW.
F-0050	ROD BLOCKS ARE RECEIVED WHEN GROUP 1 AND 2 ARE WITHDRAWN TO 48.

Evaluators: C. Persson, B. Panfil, J. Steward, B. Stetson, J. McHugh

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Test T.4.3.2, REACTOR CRITICAL TO TURBINE SYNCHRONIZED
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (02) Normal Plant Evolutions; Nuclear Startup from Hot Standby to Rated Power

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Date Tested: 05/25/91

Run Time: 16.65 hours

Test Description: The purpose of this test is demonstrate the simulators ability to conduct a normal plant startup and heatup from reactor criticality to generator on-line, using only operator action normal to Perry Unit 1. At the beginning of this test, the simulated plant has just completed a reactor startup to criticality. A nuclear plant heatup is performed and BOP systems are started at various Reactor Pressures in accordance with the System Operating Instructions and IOI-1. Special periodic tests are also performed on NSSS Systems such as RCIC, RC&IS, and SRV's during the startup. The Reactor Feed Pumps, Off-Gas system, Steam Seal System, and various other auxiliaries are started in sequence. At rated system pressure, the Main Turbine is warmed up and a Turbine Roll is conducted. At the conclusion of the test, the Reactor is at 12% Power, and the Main Turbine Generator is connected to the Grid supplying approximately 100 MWe.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
- IOI-1 COLD STARTUP
- SOI-B21 NUCLEAR STM SUPPLY SHUTOFF, AUTO DEPRESSURIZATION, & NSSS
- SOI-C34 FEEDWATER CONTROL SYSTEM (UNIT 1)
- SOI-C51(IRM) INTERMEDIATE RANGE MONITORING SYSTEM (UNIT 1)
- SOI-C51(SRM) SOURCE RANGE MONITORING SYSTEM (UNIT 1)
- SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
- SOI-G33 REACTOR WATER CLEANUP SYSTEM (UNIT 1)
- SOI-N11 MAIN & REHEAT STEAM SYSTEM (UNIT 1)
- SOI-N23 CONDENSATE FILTRATION SYSTEM (UNIT 1)
- SOI-N24 CONDENSATE DEMINERALIZER SYSTEM (UNIT 1)
- SOI-N27 FEEDWATER SYSTEM (UNIT 1)
- SOI-N32/39 MAIN TURBINE & TURNING GEAR SYSTEM (UNIT 1)
- SOI-N33 STEAM SEAL SYSTEM (UNIT 1)
- SOI-N64/62 OFF-GAS/CONDENSER AIR REMOVAL SYSTEM
- SOI-S11 POWER TRANSFORMERS

Reference Type: Surveillance & Periodic Test Procedures
- PTI-N32-P0001 TURB OVERSPEED PROT DEVICES TRIP & EHC/TURB LUBE OIL PUMP STARTS/STATOR WATER PUMP START, & ROTATIONS WEEKLY TEST
- SVI-B21-T2001 MSIV FULL STROKE OPERABILITY TEST
- SVI-B21-T2005 SRV EXERCISE TEST
- SVI-E51-T1272 PROC/INSTR PERIODIC REVIEW - RCIC SYSTEM LOW PRESS TEST
- SVI-E51-T2001 RCIC PUMP & VALVE OPERABILITY TEST
- SVI-N31-T1151 MAIN TURBINE VALVE EXERCISE TEST

Test T.4.3.2, REACTOR CRITICAL TO TURBINE SYNCHRONIZED

Malfunctions Tested: None

Major Discrepancies:

P-1514	IF RWCU TRIPS AT APPROXIMATELY 250#, PRESSURE AND FLOWS ARE UNSTABLE
P-1515	NUMEROUS OG PARAMETERS OUT OF BA[illegible] T WITHIN LAWS OF PHYSICS.
F-0028	IN MANUAL AND AUTO CONTROL WAS ABI[illegible] CHANGE RFPT SPEED, STEAM FI[illegible]W WITH NO CHANGE IN TURBINE CO[illegible] VALVE POSITION.
F-0059	R[illegible] LEVEL RESPONSE TO RPV HEATUP IS [illegible]APPROPRIATELY LARGE.
F-0060	RWCU FLOW OSCILLATES WITH NO CHANGE IN INPUT PARAMETERS.
F-0062	RCIC SYSTEM CANNOT ACHIEVE RATED FLOWS AND PRESSURE AT 150-200 PSI RPV PRESSURE.
F-0074	THE RHR HEAT EXCHANGERS DID NOT COOL THE SUPP POOL AT A RAPID ENOUGH RATE.
F-0080	PTS 1-5, 7-13, 15, 16, 19 ON RECORDER N31-R001 AREN'T PRINTING PROPERLY.
F-0083	THE MSL LOW PRESSURE ALARM WAS STILL IN AT 920#.
F-0086	DURING HEATUP OF MSL A/B/C/D, THE C MSL WENT TO 465 DEGREES IN SECONDS, THE OTHER 3 DO NOT INCREASE AT ALL.
F-0091	CONTROL ROD GROUP 6 FROM 00 TO 08 HAS A DRASTIC EFFECT ON REACTOR LEVEL.
F-0091	PLACING SJAE A IN SERVICE NOTED SEVERAL PROBLEMS.
F-0094	THE TURBINE PRESSURE DOES NOT DROP OFF TO 0# WHEN THE NUMBER 2 STOP VALVE PILOT IS CLOSED.
F-0115	SEE ATTACHED LIST OF CONTROLLERS (AUTO/MANUAL). WHEN CONTROLLER IS IN AUTO, THE OPEN AND CLOSE PUSHBUTTONS MOVE THE OUTPUT.

Evaluators: B. Panfil, C. Persson, B. Stetson, J. Steward, J. McHugh

Test T.4.3.3, POWER INCREASE TO 100% POWER
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (02) Normal Plant Evolutions; Nuclear Startup from Hot Standby to Rated Power

Date Tested: 05/31/91

Run Time: 8.70 hours

Test Description: The purpose of this test is to demonstrate the simulator's ability to conduct a plant power increase to 100% rated power and operator conducted surveillances, using only operator action normal to Perry Unit 1. At the beginning of this test, the main Turbine-Generator is synchronized to the grid and Reactor Power is 12%. The Unit is maneuvered in accordance with Perry Procedures IOI-1 and IOI-3 to the full power condition. The Power increase is achieved by Control Rod withdrawal and core flow increases. Supporting equipment such as additional feed pumps and MSR's are started as directed by the procedures as power is increased. Once the 100% operating condition is reached, additional sub-tests are performed to validate simulator performance. These sub-tests consist of operator conducted surveillances on safety related equipment and system operating instruction sections not performed as part of a normal startup and shutdown, such as shifting pumps, fans, controllers, etc. At the conclusion of this test, the Reactor is operating at 100% power and the Main Generator is supplying Unit rated electrical power.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B02 CONTROL ROD MOVEMENTS
IOI-1 COLD STARTUP
IOI-3 POWER CHANGES
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
SOI-C11(RCIS) CONTROL ROD & INFORMATION SYSTEM (UNIT 1)
SOI-C22 REDUNDANT REACTIVITY CONTROL SYSTEM (UNIT 1)
SOI-C34 FEEDWATER CONTROL SYSTEM (UNIT 1)
SOI-C51(APRM) AVERAGE POWER RANGE MONITORING SYSTEM (UNIT 1)
SOI-C85 STEAM BYPASS & PRESSURE REGULATOR SYSTEM (UNIT 1)
SOI-D21 AREA RADIATION MONITORING SYSTEM (D21)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E21 LOW PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E22B DIVISION 3 DIESEL GENERATOR (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-G33 REACTOR WATER CLEANUP SYSTEM (UNIT 1)
SOI-G36 RWCU FILTER/ DEMINERALIZER SYSTEM (UNIT 1)
SOI-G41(FPCC) FUEL POOL COOLING & CLEANUP SYSTEM (UNIT 1)
SOI-G42 SUPPRESSION POOL CLEANUP SYSTEM (UNIT 1)
SOI-M11 CONTAINMENT VESSEL COOLING SYSTEM (UNIT 1)
SOI-M13 DRYWELL COOLING SYSTEM (UNIT 1)
SOI-M14 CONTAINMENT VESSEL & DRYWELL PURGE SYSTEM (UNIT 1)

Test T.4.3.3, POWER INCREASE TO 100% POWER

SOI-M15 ANNULUS EXHAUST GAS TREATMENT SYSTEM (UNIT 1)
SOI-M21 CONTROLLED ACCESS & MISC EQUIPMENT AREA (CA&MEA) HVAC SYSTEMS
SOI-M23/24 MCC, SWGR, & MISC ELECTRICAL EQUIPMENT AREA HVAC SYSTEMS
SOI-M25/26 CONTROL ROOM HVAC & EMERGENCY RECIRCULATION SYSTEM
SOI-M28 EMERGENCY CLOSED COOLING PUMP AREA COOLING SYSTEM
SOI-M32 EMERGENCY SERVICE WATER PUMP HOUSE VENTILATION SYSTEM (UNIT 1)
SOI-M36 OFF-GAS BUILDING EXHAUST SYSTEM (UNIT 1)
SOI-M39 ECCS PUMP ROOMS COOLING SYSTEM (UNIT 1)
SOI-M40 FUEL HANDLING BUILDING VENTILATION SYSTEM
SOI-M43 DIESEL GENERATOR BUILDING VENTILATION SYSTEM (UNIT 1)
SOI-M51/56 COMBUSTIBLE GAS CONTROL SYSTEM & HYDROGEN IGNITERS (UNIT 1)
SOI-N11 MAIN & REHEAT STEAM SYSTEM (UNIT 1)
SOI-N21 CONDENSATE SYSTEM (UNIT 1)
SOI-N23 CONDENSATE FILTRATION SYSTEM (UNIT 1)
SOI-N24 CONDENSATE DEMINERALIZER SYSTEM (UNIT 1)
SOI-N27 FEEDWATER SYSTEM (UNIT 1)
SOI-N32/39 MAIN TURBINE & TURNING GEAR SYSTEM (UNIT 1)
SOI-N33 STEAM SEAL SYSTEM (UNIT 1)
SOI-N41/51 MAIN GENERATOR & EXCITATION SYSTEM (UNIT 1)
SOI-N42 HYDROGEN SEAL OIL SYSTEM (UNIT 1)
SOI-N43 STATOR WATER COOLING SYSTEM (UNIT 1)
SOI-N64/62 OFF-GAS/CONDENSER AIR REMOVAL SYSTEM
SOI-N71 CIRCULATING WATER/CONDENSER MECHANICAL CLEANING SYSTEM (UNIT 1)
SOI-P11 CONDENSATE TRANSFER & STORAGE SYSTEM (UNIT 1)
SOI-P42 EMERGENCY CLOSED COOLING SYSTEM (UNIT 1)
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)
SOI-P47 CONTROL COMPLEX CHILLED WATER SYSTEM
SOI-P51/52 SERVICE & INSTRUMENT AIR SYSTEM
SOI-R10 PLANT ELECTRICAL SYSTEM
SOI-R13 ISOLATION PHASE BUS DUCT COOLING SYSTEM (UNIT 1)
SOI-R14 120V AC VITAL INVERTERS (UNIT 1)
SOI-R42(DIV 1) DIV 1 DC DISTR BUSES ED-1-A & ED-2-A: BATTERIES, CHARGERS, & SWITCHGEAR
SOI-R42(DIV 2) DIV 2 DC DISTR BUSES ED-1-B & ED-2-B: BATTERIES, CHARGERS, & SWITCHGEAR
SOI-R42(DIV 3) DIV 3 DC DISTR BUSES ED-1-C & ED-2-C: BATTERIES, CHARGERS, & SWITCHGEAR
SOI-R42(SYS A) NON DIV DC SYS A DISTR BUSES D-1-A & D-2-A: BATTERIES, CHARGERS, & SWITCHGEAR
SOI-R42(SYS B) NON DIV DC SYS B DISTR BUSES D-1-B & D-2-B: BATTERIES, CHARGERS, & SWITCHGEAR
SOI-R43 DIVISION 1 & 2 DIESEL GENERATOR SYSTEM (UNIT 1)

Reference Type: Surveillance & Periodic Test Procedures
PTI-N27-P0001 REACTOR FEED PUMP TURBINE STOP VALVE TEST
PTI-N27-P0002 REACTOR FEED PUMP TURBINES THRUST BEARING WEAR TEST
PTI-N27-P0005 RFP TURBINES LOCKOUT SUPPRESSED OVERSPEED TRIP TEST
PTI-N27-P0006 EMERGENCY LUBE OIL PUMP TEST FOR MOTOR DRIVEN FEED PUMP
PTI-N32-P0002 TURBINE - GENERATOR OIL SYSTEMS MONTHLY TESTING
PTI-N36-P0001 WEEKLY NON-RETURN CHECK VALVE TEST
SVI-B33-T1160 JET PUMP OPERABILITY
SVI-C11-T0009 CONTROL ROD SCRAM ACCUMULATOR PRESS/LEAK DETECTION TESTS

Test T.4.3.3, POWER INCREASE TO 100% POWER

SVI-C11-T1003 CONTROL ROD EXERCISE
SVI-C11-T1022 ROD PATTERN CONTROL SYSTEM - ROD WITHDRAWAL LIMITER
SVI-C41-T2001 STANDBY LIQUID CONTROL PUMP & VALVE OPERABILITY TEST
SVI-C51-T0024 APRM GAIN & CHANNEL CALIBRATION
SVI-C85-T1314 TURBINE BYPASS VALVE OPERABILITY TEST
SVI-E22-T1319 DIESEL GENERATOR START & LOAD DIVISION
SVI-M26-T1259 CONTROL ROOM EMERGENCY RECIRCULATION OPERABILITY TEST
SVI-M51-T2003 COMBUSTIBLE GAS MIXING SYSTEM OPERABILITY TEST
SVI-R10-T5217 ELECTRICAL DISTRIBUTION SYSTEM ENERGIZATION CHECK
SVI-R43-T1317 DIESEL GENERATOR START & LOAD DIVISION 1

Malfunctions Tested: None

Major Discrepancies:

F-0119	CAN'T GET JET PUMP OPERABILITY SVI-B33-T1160 TO PASS.
F-0126	COULD NOT TRANSFER B33-K603B TO MANUAL, AFTER IT WAS IN AUTO.
F-0127	COULD NOT GET THE A LOOP FLOW CONTROL FOR REACTOR RECIRC ON THE AUTO CONTROLLER WITHOUT A 5K LOOP FLOW MISMATCH.
F-0128	IOI-1 REQUIRES FEEDWATER TEMPERATURE TO BE GREATER THAN PDB-A011. THIS IS AT 325 DEG F, SIMULATOR SHOWED 292 DEG F.
F-0130	WHILE SHIFTING FROM RX FEED PUMP A ON THE S/U LEVEL CONTROLLER TO THE MASTER LEVEL CONTROLLER, NOTED A SEVERE BUMP OCCURRED.
F-0133	THE OUTER SEALS ON BOTH REACTOR RECIRC PUMPS ARE FAILED.
F-0139	OSCILLATIONS IN TURBINE 1ST AND 2ND STAGE REHEAT FLOW STM FLOW OCCUR WHENEVER TURBINE LOAD IS CHANGED.
F-0141	PTV 1P41F003 IS NOT RESPONDING TO MAINTAIN TBCC TEMPERATURE.
F-0144	COULD NOT BRING ON ONE REACTOR WATER CLEANUP FILTER AT A TIME. BOTH CAME UP TOGETHER WHEN B WAS PLACED ON LINE.

Evaluators: B. Panfil, C. Persson, J. Steward, J. McHugh

Test T.4.3.4, POWER DECREASE TO TURBINE/GENERATOR UNLOADED
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (08) Normal Plant Evolutions; Plant Shutdown from Rated Power to Hot Standby

Date Tested: 06/10/91

Run Time: 8.00 hours

Test Description: The purpose of this test is to demonstrate the simulator's ability to conduct a normal plant shutdown from 100% power to generator off-line, using only operator action normal to Perry Unit 1. At the beginning of this test, the simulated plant is operating at rated 100% Power capacity. The test consists of performing a normal reactor plant power decrease per Operating Procedures IOI-3, IOI-4 and supporting SOI's contained in the PNPP Unit 1 Operations Manual. Reactor Power is decreased by inserting control rods and lowering reactor recirculation core flow. Supporting plant systems are similarly maneuvered per the IOI. The turbine is completely unloaded and shutdown. At the completion of the test, Reactor Power is approximately 8%, steam is bypassed to the Main Condenser, and the turbine is coasting down from 1800 RPM.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
- FTI-B02 CONTROL ROD MOVEMENTS
- IOI-3 POWER CHANGES
- IOI-4 SHUTDOWN
- SOI-B33 REACTOR RECIRCULATION SYSTEM
- SOI-C11(RCIS) CONTROL ROD & INFORMATION SYSTEM (UNIT 1)
- SOI-C34 FEEDWATER CONTROL SYSTEM (UNIT 1)
- SOI-C51(IRM) INTERMEDIATE RANGE MONITORING SYSTEM (UNIT 1)
- SOI-G33 REACTOR WATER CLEANUP SYSTEM (UNIT 1)
- SOI-N11 MAIN & REHEAT STEAM SYSTEM (UNIT 1)
- SOI-N21 CONDENSATE SYSTEM (UNIT 1)
- SOI-N23 CONDENSATE FILTRATION SYSTEM (UNIT 1)
- SOI-N24 CONDENSATE DEMINERALIZER SYSTEM (UNIT 1)
- SOI-N27 FEEDWATER SYSTEM (UNIT 1)
- SOI-N32/39 MAIN TURBINE & TURNING GEAR SYSTEM (UNIT 1)
- SOI-R10 PLANT ELECTRICAL SYSTEM
- SOI-R13 ISOLATION PHASE BUS DUCT COOLING SYSTEM (UNIT 1)
- SOI-S11 POWER TRANSFORMERS

Reference Type: Surveillance Procedures
- SVI-C11-T1019 ROD PATTERN CONTROLLER SYS TEST BELOW LOW POWER SETPOINT
- SVI-C11-T1022 ROD PATTERN CONTROL SYSTEM - ROD WITHDRAWAL LIMITER

Test T.4.3.4, POWER DECREASE TO TURBINE/GENERATOR UNLOADED

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.3.5, PLANT COOLDOWN TO COLD SHUTDOWN
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (08) Normal Plant Evolutions; Shutdown and Cooldown to Cold Shutdown Conditions

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Date Tested: 06/10/91

Run Time: 5.50 hours

Test Description: The purpose of this test is to demonstrate the simulator's ability to conduct a normal reactor plant shutdown and cooldown from generator off-line to the cold shutdown condition (<200 deg F), using only operator action normal to Perry Unit 1. At the beginning of this test, the reactor is at about 8% power. Control rods are inserted to complete the reactor shutdown. Auxiliary BOP systems are shutdown as instructed by the Operating Procedure IOI-4. After the reactor is shutdown, the plant is lined up to support a plant cooldown. Steam is dumped to the main condenser and the cooldown is controlled <100 deg F/hr as it is in the reference plant. The RHR systems are flushed and prepared to support the shutdown cooling mode of operation. At the prescribed temperature, cooldown is shifted to the RHR system, and the steam plant is shutdown. Cooldown is continued with RHR until the reactor temperature is in the normal cold shutdown band of 120-140 deg F. At the test conclusion, the steam systems are shutdown, vacuum in the main condenser has been broken, RRC pumps are secured, and temperature is being controlled by RHR in SDC mode.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
- FTI-B02 CONTROL ROD MOVEMENTS
- IOI-12 MAINTAINING COLD SHUTDOWN
- IOI-4 SHUTDOWN
- SOI-B21 NUCLEAR STM SUPPLY SHUTOFF, AUTO DEPRESSURIZATION, & NSSS
- SOI-B33 REACTOR RECIRCULATION SYSTEM
- SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
- SOI-C34 FEEDWATER CONTROL SYSTEM (UNIT 1)
- SOI-C51(SRM) SOURCE RANGE MONITORING SYSTEM (UNIT 1)
- SOI-C71 RPS POWER SUPPLY DISTRIBUTION (UNIT 1)
- SOI-C85 STEAM BYPASS & PRESSURE REGULATOR SYSTEM (UNIT 1)
- SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
- SOI-G33 REACTOR WATER CLEANUP SYSTEM (UNIT 1)
- SOI-N21 CONDENSATE SYSTEM (UNIT 1)
- SOI-N24 CONDENSATE DEMINERALIZER SYSTEM (UNIT 1)
- SOI-N27 FEEDWATER SYSTEM (UNIT 1)
- SOI-N33 STEAM SEAL SYSTEM (UNIT 1)
- SOI-N64(OGVRS) OFF-GAS VAULT REFRIGERATION SYSTEM (UNIT 1)
- SOI-N64/62 OFF-GAS/CONDENSER AIR REMOVAL SYSTEM
- SOI-P12 CONDENSATE SEAL SYSTEM (UNIT 1)

Test T.4.3.5, PLANT COOLDOWN TO COLD SHUTDOWN

SOI-P42 EMERGENCY CLOSED COOLING SYSTEM (UNIT 1)
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)
SOI-P54 FIRE PROTECTION SYSTEM

Reference Type: Surveillance Procedures
SVI-B21-T1176 OM7A SVI RCS HEATUP & COOLDOWN SURVEILLANCE

Malfunctions Tested: None

Major Discrepancies:

P-0125	STEPS 28-35 OF T.4.3.5 MUST BE REPEATED. RHR INDICATIONS FOR FLUSH PRIOR TO INITIATING SDC MODE IMPROPER.
F-0190	WHILE PERFORMING HEAD SPRAY SHELL AND HEAD FLANGE TEMPS ARE 100 DEGREES LESS THAN BOTTOM DRAIN TEMPS.
F-0191	WHILE FILLING RX VESSEL TO 490-500", COULDN'T FILL ABOVE 400" WITHOUT CAUSING RPV PRESS TO INCREASE.
F-0193	WHILE DECREASING PRESSURE WITH BYPASS VALVES A 10# PRESS DECREASE CAUSES 25" TO 30" LEVEL INCREASE (SWELL).
F-0198	WHILE PERFORMING RHR S/D COOLING FLUSH WAS UNABLE TO MAINTAIN RHR HEADER PRESS WITH MINIMAL FLOW THROTTLE POSITION (F040).
F-0199	UNABLE TO OBTAIN PROPER FLOW RATE (6900-7100) WHILE IN SUPPRESSION POOL COOLING WITH ALL SYSTEM FLOW THROUGH RHR A HEAT EXCHANGERS.
F-0201	REMOTE FUNCTION RH24 (1E12-F315) NEEDS TO HAVE EITHER THROTTLING OR MANUAL OPERATION CAPABILITY TO ALLOW FOR VESSEL FEED AT AN ADJUSTABLE RATE TO SUPPORT COLD SHUTDOWN OPERATION.
F-0208	WHEN ATTEMPTING TO FEED THE VESSEL WITH HOTWELL PUMPS THROUGH THE CHEMICAL CLEANING LINE, CONDENSATE FLOW READS 10,000 GPM, BUT NO LEVEL INCREASE WAS OBSERVED.

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

ATP TEST SECTION T.4.4.1

STEADY STATE PERFORMANCE

Overview of testing performed in this section

This series of tests verifies the critical parameters for the Perry Unit 1 Simulator to be in compliance with the accuracy requirements of Section 4 of the Standard for the steady state heat balances. This section also includes the tests for steady state stability verification.

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.4.1.1, 25% POWER HEAT BALANCE	2	(N/C)*
T.4.4.1.2, 50% POWER HEAT BALANCE	3	(N/C)*
T.4.4.1.3, 75% POWER HEAT BALANCE	4	(N/C)*
T.4.4.1.4, 100% POWER HEAT BALANCE	5	(N/C)*
T.4.4.1.5, 100% POWER STABILITY TEST	6	(N/C)*

ANSI/ANS 3.5 Reference: *Appendix B 1.1 Steady State Performance*

* This test has not been completed (N/C) or has been performed but has one or more unresolved major Simulator Discrepancy Reports associated with it. Please see the individual test abstract for these discrepancies which constitute exceptions to ANSI/ANS 3.5 section 3.1.

Test T.4.4.1.1, 25% POWER HEAT BALANCE
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.2 Steady State Performance

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Date Tested: Not RUN

Run Time: Later hours

Test Description: This test verifies the critical parameters for the Perry Unit 1 Simulator to be in compliance with the accuracy requirements of Section 4 of the Standard at a power level of 25%. The simulator is initialized to 25% power, slow speed pumps. Various primary and secondary plant parameters are recorded and compared to design and actual plant values, where available.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B05 CORE HEAT BALANCE

Reference Type: Other
GE Thermal Kit BALANCE OF PLANT THERMAL KIT (4 MS/R VESSELS)

Reference Type: Plant Data
DR-176-S GE REACTOR SYSTEM HEAT BALANCE - RATED
Heat Balance SPECIAL LOG SERVICES MODIFY SPECIAL LOG 9 FT1 B-05 HEAT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.4.1.2, 50% POWER HEAT BALANCE
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.2 Steady State Performance

Date Tested: Not RUN

Run Time: Later hours

Test Description: This test verifies the critical parameters for the Perry Unit 1 Simulator to be in compliance with the accuracy requirements of Section 4 of the Standard at a power level of 50%. The simulator is initialized to 50% power, fast speed pumps. Various primary and secondary plant parameters are recorded and compared to design and actual plant values, where available.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B05 CORE HEAT BALANCE

Reference Type: Other
GE Thermal Kit BALANCE OF PLANT THERMAL KIT (4 MS/R VESSELS)

Reference Type: Plant Data
DR-176-S GE REACTOR SYSTEM HEAT BALANCE - RATED
Heat Balance SPECIAL LOG SERVICES MODIFY SPECIAL LOG 9 FT1 B-05 HEAT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.4.1.3, 75% POWER HEAT BALANCE
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.2 Steady State Performance

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Date Tested: Not RUN

Run Time: Later hours

Test Description: This test verifies the critical parameters for the Perry Unit 1 Simulator to be in compliance with the accuracy requirements of Section 4 of the Standard at a power level of 75%. The simulator is initialized to 75% power, fast speed pumps. Various primary and secondary plant parameters are recorded and compared to design and actual plant values, where available.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B05 CORE HEAT BALANCE

Reference Type: Other
GE Thermal Kit BALANCE OF PLANT THERMAL KIT (4 MS/R VESSELS)

Reference Type: Plant Data
DR-176-S GE REACTOR SYSTEM HEAT BALANCE - RATED
Heat Balance SPECIAL LOG SERVICES MODIFY SPECIAL LOG 9 FT1 B-05 HEAT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.4.1.4, 100% POWER HEAT BALANCE
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.2 Steady State Performance

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Date Tested: Not RUN

Run Time: Later hours

Test Description: This test verifies the critical parameters for the Perry Unit 1 Simulator to be in compliance with the accuracy requirements of Section 4 of the Standard at a power level of 100%. The simulator is initialized to 100% power, fast speed pumps. Various primary and secondary plant parameters are recorded and compared to design and actual plant values, where available.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B05 CORE HEAT BALANCE

Reference Type: Other
GE Thermal Kit BALANCE OF PLANT THERMAL KIT (4 MS/R VESSELS)

Reference Type: Plant Data
DR-176-S GE REACTOR SYSTEM HEAT BALANCE - RATED
Heat Balance SPECIAL LOG SERVICES MODIFY SPECIAL LOG 9 FT1 B-05 HEAT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.4.1.5, 100% POWER STABILITY TEST
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.2 Steady State Performance

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Date Tested: Not RUN

Run Time: Later hours

Test Description: This test verifies the stability of the simulation at 100% power level. The simulator is initialized to 100% steady state power IC. The simulator is run for a minimum of one hour at a constant power level with no operator or instructor actions performed. The value of critical parameters and various other parameters are recorded throughout the evolution and verified to meet the criteria of the Standard for stability.

Baseline Data used For Reference

Reference Type: Other
Simulator Specification

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

ATP TEST SECTION T.4.4.2

TRANSIENT PERFORMANCE

Overview of testing performed in this section

This series of tests verifies the Perry Unit 1 Simulator to be in compliance with the transient criteria of Section 4 of the Standard for the 10 benchmark transients described in Appendix B of the Standard.

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.4.2.01, MANUAL SCRAM	2	None
T.4.4.2.02, SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS	3	1*
T.4.4.2.03, SIMULTANEOUS CLOSURE OF ALL MSIV'S	4	None
T.4.4.2.04, SIMULTANEOUS TRIP OF ALL RECIRC PUMPS	5	None
T.4.4.2.05, SINGLE RECIRC PUMP TRIP	6	None
T.4.4.2.06, MAIN TURBINE TRIP W/O RX SCRAM	7	None
T.4.4.2.07, MAXIMUM RATE POWER RAMP (100% - 75% - 100%)	8	1*
T.4.4.2.08, MAXIMUM SIZE LOCA W/LOSS OFFSITE POWER	9	(N/C)*
T.4.4.2.09, MAXIMUM SIZE UNISOLABLE MSL RUPTURE	10	(N/C)*
T.4.4.2.10, SIMULTANEOUS CLOSURE OF MSIV'S W/SINGLE STUCK OPEN SAFETY/RELIEF VALVE	11	2*

ANSI/ANS 3.5 Reference: *Appendix B 1.2* *Transient Performance*

* This test has not been completed (N/C) or has been performed but has one or more unresolved major Simulator Discrepancy Reports associated with it. Please see the individual test abstract for these discrepancies which constitute exceptions to ANSI/ANS 3.5 section 3.1.

Test T.4.4.2.01, MANUAL SCRAM
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/15/91

Run Time: 1.5 hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a manual Reactor Scram from Full Power. The simulator is initialized to 100% Power IC. A manual scram is performed. Critical parameters are recorded and analyzed for comparison to plant data. At the end of the test, the reactor is stabilized in a post-scram condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-C71-1 REACTOR SCRAM (UNIT 1)

Reference Type: Other
Best Estimate

Reference Type: Plant Data
SER-1-88-5 SCRAM EVALUATION REPORT
SER-1-88-7 SCRAM EVALUATION REPORT

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.4.2.02, SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/15/91

Run Time: 1 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a simultaneous trip of all normal feedwater pumps. The simulator is initialized to 100% steady state power IC. The Motor Feed Pump is placed in 'OFF' and both running RFPT's are tripped. Critical parameter are recorded per the ANS list, and compared to plant data. At the end of the test, the plant is stabilized in the post-scram condition with RPV level being maintained by HPCS and RCIC systems.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N27 FEEDWATER PUMP TRIP (UNIT 1)

Reference Type: Other
Best Estimate

Reference Type: Plant Data
SER-1-88-2 SCRAM EVALUATION REPORT
SER-1-88-5 SCRAM EVALUATION REPORT

Reference Type: Plant Data - Analyses
USAR 15.2.7 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: None

Major Discrepancies:

F-0239 DURING LOSS OF FW ACCIDENT, FOLLOWING HPCS/RCIC L8 SHUTOFF, APPROX. 100" OF INDICATED LEVEL IS "LOST" IN ~1 MIN. LEVEL SHOULD INCREASE

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.4.2.03, SIMULTANEOUS CLOSURE OF ALL MSIV'S
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/15/91

Run Time: 1 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a simultaneous closure of all MSIV's. The simulator is initialized to 100% power IC. The MSIV's are simultaneously closed. Critical parameters are recorded and compared to expected results for validation.

Baseline Data used For Reference

Reference Type: Plant Data - Analyses
USAR 15.2.4 PNPP UPDATED SAFETY ANALYSIS REPORT

Reference Type: Plant Data - Startup Test Results
STI-B21-025B VESSEL ISOLATION

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.4.2.04, SIMULTANEOUS TRIP OF ALL RECIRC PUMPS
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/16/91

Run Time: 1 hour

Test Description: The purpose of this test is to verify simulator response to a simultaneous trip of both Reactor Recirculation Pumps. The simulator is initialized to 100% steady state power IC with recirc pumps in fast speed. Both pumps are tripped simultaneously while critical parameters are recorded. The plant stabilizes in a post-Scram condition with natural recirculation core flow.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-B33-2 LOSS OF ONE OR BOTH RECIRCULATION PUMPS (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.3.1 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.4.2.05, SINGLE RECIRC PUMP TRIP
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/16/91

Run Time: 1.5 hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a single Reactor Recirculation Pump trip. The simulator is initialized to 100% Steady State power IC. The selected pump is tripped while critical parameters are recorded. Parameter response is compared to plant and USAR data for validation of response.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-B33-1 REACTOR RECIRCULATION FLOW CONTROL MALFUNCTION (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.3.1 PNPP UPDATED SAFETY ANALYSIS REPORT

Reference Type: Plant Data - Startup Test Results
STI-B33-030A/2 DATA REQUEST RESPONSE - ONE RECIRC PUMP TRIP & RESTART

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.4.2.06, MAIN TURBINE TRIP W/O RX SCRAM
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/16/91

Run Time: 0.5 hour

Test Description: The purpose of this test is to verify the proper response of the Simulator to a main turbine trip initiated from a power level which does not result in a Reactor Scram. Initial conditions: The simulator is initialized to 30% steady state power. The main turbine is tripped. Turbine bypass valves are verified to open to maintain reactor pressure constant. Critical parameters are recorded and compared to analysis and best estimate data. At the completion of the test, the reactor is operating at about 30% power with bypass valves open.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.2.3 PNPP UPDATED SAFETY ANALYSIS REPORT

Reference Type: Plant Data - Startup Test Results
STI-B21-027/2 DATA REQUEST RESPONSE - TURB TRIP & GEN LD REJ

Malfunctions Tested: None

Major Discrepancies: None

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

Test T.4.4.2.07, MAXIMUM RATE POWER RAMP (100% - 75% - 100%)
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/15/91

Run Time: 0.7 hour

Test Description: The purpose of this test is to verify Simulator response to power ramps initiated by changes in Reactor Recirc flow. The simulator is initialized to 100% Steady State power IC. Using the normal mode of recirc flow control, reactor power is reduced at the maximum rate to 75% power, then increased at the maximum rate to 100% power. Critical parameters are recorded and analyzed to validate the response. Plant parameters at the end of the test are the same as initial, a xenon transient is expected.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Reference Type: Plant Data - Startup Test Results
STI-B33-029B/3 DATA REQUEST RESPONSE - RECIRCULATION FLOW CONTROL
STI-B33-030C/4 DATA REQUEST RESPONSE - RECIRCULATION SYSTEM

Malfunctions Tested: None

Major Discrepancies:

F-0236 IF PWR IS REDUCED TO 75% A MOMENTARY STM SEAL EVAP DNTK LVL HI SIGNAL INAPPROPRIATELY ISOLATES EXTRACTION STEAM TO SSE CAUSING A DECREASE IN SUPPLY TO STEAM SEAL HEADERS AND HDR PRESS IS LOST.

Evaluators: B. Panf[illegible] [illegible] Persson, G. Minshall, H. DeBoer

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Test T.4.4.2.08, MAXIMUM SIZE LOCA W/LOSS OF OFFSITE POWER
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: Not Completed

Run Time: Later hours

Test Description: The purpose of this test is to verify proper response of the Simulator to a DBA LOCA concurrent with a loss of Off-Site Power. Initial Conditions: The simulator is initialized to 100% power steady state IC. A simultaneous Recirc Loop piping rupture and Loss of off-site power is initiated while recording critical parameters. The transient is allowed to continue until a somewhat stabilized condition is attained. At the end of the test, the reactor is shutdown with control rods inserted, the reactor vessel is depressurized, Low Pressure ECCS is injecting to provide adequate core cooling. Containment parameters are out of their normal ranges, but are verified to not have exceeded any design limitations.

[illegible] Data used For Reference

Reference Type: Off-Normal Operating Procedures
[illegible]I-S11 LOSS OF OFF-SITE POWER

Reference Type: Other
Best Estimate

Reference Type: Plant Data - Analyses
USAR 15.6.5 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: TH01 RECIRC LOOP RUPTURE (DBA LOCA)

Discrepancies: Unknown

Evaluators: Later

Test T.4.4.2.09, MAXIMUM SIZE UNISOLABLE MSL RUPTURE
Revision Number: Later
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: Not Completed

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response to a design basis Steam Line Rupture Accident in the Drywell. Initial Conditions: The simulator is initialized to 100% Steady State Power IC. Malfunction TH26 is activated. Critical parameters are recorded and compared to analysis and best estimate data. At the end of the test, the reactor is shutdown and depressurized, low pressure ECCS has flooded the RPV to the level of the break to provide adequate core cooling; Containment parameters are out of normal ranges, but are verified to not have exceeded design limits.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Reference Type: Plant Data - Analyses
USAR 15.6.5 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: TH26 MAIN STEAM LINE RUPTURE INSIDE DRYWELL

Discrepancies: Unknown

Evaluators: Later

Test T.4.4.2.10, SIMULTANEOUS CLOSURE OF MSIV'S W/SINGLE STUCK OPEN SAF
Revision Number: 00
ANSI/ANS 3.5 Section: Appendix A 3.3 Transient Tests

Date Tested: 06/16/91

Run Time: 1.5 hours

Test Description: The purpose of this test is to verify proper Simulator response to a simultaneous closure of the MSIV's with a stuck open SRV and no high pressure injection available. Initial conditions: the simulator is initialized to 100% power IC. Prior to activating the transient, the HPCS and RCIC systems are disabled, and the Motor Feed Pump is placed in secured status. One SRV is overridden open. The MSIV's are closed and critical parameters are recorded until a somewhat stabilized condition is obtained. At the completion of the test, the reactor is shutdown and depressurized due to ADS initiation. Low pressure ECCS pumps are injecting to the RPV to maintain adequate core cooling. Containment parameters are out of normal ranges, but are verified to not exceed design limits.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Reference Type: Plant Data - Startup Test Results
STI-B21-025B VESSEL ISOLATION
STI-B21-026/1 DATA REQUEST RESPONSE - SAFETY RELIEF VALVES

Malfunctions Tested: None

Major Discrepancies:

F-0229 WITH LOCA SIGNAL PRESENT SUPP POOL MAKEUP VLVS AUTO OPEN UPON SENSING LO SP LVL ALARM SETPT(18-18.1'). SHOULD OPEN AT LEVEL <16.75' W/LOCA.

F-0230 UPON AUTO INITIATION OF LPCS DUE TO LOW RPV LEVEL, THE LPCS PUMP MOTOR STARTED AND THEN IMMEDIATELY TRIPPED OFF DUE TO OVERCURRENT.

Evaluators: B. Panfil, C. Persson, G. Minshall, H. DeBoer

ATP TEST SECTION T.4.5.1

SYSTEM LEVEL FAILURES

Overview of testing performed in this section

This series of Tests was performed to verify the proper response of the Simulator to the set of System Level Failures provided in the design of the Simulator. The test methodology was to first verify proper "first principles" modeling of the simulated failure on the system level, then to verify the proper response of any interfacing systems in the integrated environment. Finally, each failure was to be verified to provide the proper overall plant response as predicted for the particular type of failure. Test procedures were written to check each of these items. These tests were written by Utility (CEI) employees holding a current SRO license on the Perry Unit 1 Plant and have operational and training experience. (These personnel were on loan to the Simulator Vendor from July 1989, and personally observed or performed all Operational Testing of the Simulator.) The test abstracts include a description of which generic cases were tested for each generic malfunction, and the severities which were tested for those having adjustable rates. Please refer to the Failure Cause and Effects Manual (available at Perry) for a complete description of the failure modes, the generic cases available, and the range of severities applicable to each System Level Failure.

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.5.1.02.AN01, ANNUNCIATOR INPUT OPTICAL ISOLATOR FAILURE	3	2*
T.4.5.1.04.CU03, RWCU SYSTEM PIPE BREAK OUTSIDE CONTAINMENT (STEAM TUNNEL)	4	2*
T.4.5.1.06.ED06, LOSS OF 480V BUS	5	None
T.4.5.1.06.ED09, LOSS OF 125 VDC BUS	6	6*
T.4.5.1.06.ED17, LOSS OF 125V DC DISTRIBUTION PANEL	8	(N/C)*
T.4.5.1.07.EG01, MAIN GENERATOR LOCKOUT RELAY TRIP	9	None
T.4.5.1.09.FW02, FEEDWATER SYSTEM PIPE BREAK INSIDE DRYWELL	10	4*
T.4.5.1.09.FW03, FEEDWATER SYSTEM PIPE BREAK OUTSIDE CONTAINMENT	11	4*
T.4.5.1.12.IA01, AIR RECEIVER LEAK	12	9*
T.4.5.1.12.IA02, INSTRUMENT AIR LINE LEAK	14	6*
T.4.5.1.15.MC01, CONDENSER AIR INLEAKAGE	15	(N/C)*
T.4.5.1.16.MS11, STEAM SEAL HEADER PRESSURE REGULATOR FAILURE	16	None
T.4.5.1.17.NM01, SRM DETECTOR (PRE-AMP) FAILURE	17	1*
T.4.5.1.17.NM02, IRM DETECTOR (PRE-AMP) FAILURE	18	None
T.4.5.1.17.NM03, LPRM DETECTOR FAILURE	19	None
T.4.5.1.17.NM10, NEUTRON MONITORING DETECTOR DRIVE STUCK	20	None
T.4.5.1.18.OG03, OFF GAS SYSTEM LEAK UPSTREAM OF ADSORBERS	21	3*
T.4.5.1.18.OG04, OFF GAS SYSTEM LEAK DOWNSTREAM OF ADSORRERS	22	None

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.5.1.21.RD03, CONTROL ROD DRIFT - IN	23	(N/C)*
T.4.5.1.21.RD04, CONTROL ROD DRIFT - OUT	24	(N/C)*
T.4.5.1.22.RH02, RESIDUAL HEAT REMOVAL SYSTEM PIPE BREAK	25	(N/C)*
T.4.5.1.23.RP02, INADVERTENT INITIATION OF ALTERNATE ROD INJECTION	26	None
T.4.5.1.25.SW01, NUCLEAR CLOSED COOLING SYSTEM PROCESS PIPING LEAKAGE	27	1*
T.4.5.1.25.SW02, SERVICE WATER SYSTEM PROCESS PIPING LEAKAGE	28	2*
T.4.5.1.27.TH01, RECIRC LOOP RUPTURE (DBA LOCA)	29	(N/C)*
T.4.5.1.27.TH19, RPV LEVEL INST REFERENCE LEG BREAK	30	1*
T.4.5.1.27.TH20, RPV LEVEL INST VARIABLE LEG BREAK	31	(N/C)*
T.4.5.1.27.TH21, POWER/FLOW INSTABILITIES (IEB 88-07 SUPPLEMENT 1)	32	1*
T.4.5.1.27.TH27, MAIN STEAM LINE RUPTURE IN STEAM TUNNEL	33	2*
T.4.5.1.28.TU01, MAIN SHAFT OIL PUMP DEGRADATION	34	None

ANSI/ANS 3.5 Reference: *3.1.2 Plant Malfunctions*

* This test has not been completed (N/C) or has been performed but has one or more unresolved major Simulator Discrepancy Reports associated with it. Please see the individual test abstract for these discrepancies which contitute exceptions to ANSI/ANS 3.5 section 3.1.2.

Test T.4.5.1.02.AN01, ANNUNCIATOR INPUT OPTICAL ISOLATOR FAILURE
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (22)(b) Plant Malfunctions; Process Alarm System Failure

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Date Tested: 05/06/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to annunciator failures. This failure simulates a loss of instrument power to annunciator system optical isolator units on the input (system) side. The failure has 39 generic cases representing a variety of isolators in numerous safety related systems (isolators are not required in BOP systems). The test is initiated from 100% power operating condition. During the test, the failure is randomly selected by the test personnel. Proper response is checked by driving the affected system parameters/conditions to the alarm setpoint/condition, and observing that the affected annunciators are not received. Annunciation of alarms identifying the power loss are verified to annunciate. The failure is deleted, simulating repairs to the isolator have been completed, and the receipt of appropriate alarms is verified. At the completion of the test the plant remains at 100% power condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
OM6: ARI PERRY UNIT ALARM RESPONSE INSTRUCTIONS

Reference Type: Other
Best Estimate

Malfunctions Tested: AN01 Annunciator Input Optical Isolator Failure

Major Discrepancies:

P-1350 MALFUNCTION AN01 CASE Z HAD NO EFFECT ON THE ANNUNCIATORS DRIVEN BY THE OPTICAL ISOLATOR. (1E22AT4).

P-1351 CERTAIN ALARMS WERE NOT RECEIVED DURING THE RUNNING OF AN01 MALFUNCTION, CASES T, V, W, AB, AC, AE, AG.

Evaluators: R.Libra, C. Persson, B. Stetson

Test T.4.5.1.04.CU03, RWCU SYSTEM PIPE BREAK OUTSIDE CONTAINMENT (STM TUNNEL)
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (01)(b) Plant Malfunctions; LOCA outside containment

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Date Tested: 06/19/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify proper simulator response to a reactor water cleanup (RWCU) pipe break outside the containment (steam tunnel). The test is initiated from 100% power operating condition. Alarms, isolation valves and positions, recorders, timers, and temperature recorder and alarms are verified to operate properly. At the end of the test, the plant is shutdown with the MSIV's and Main Steam Line Drain Valves closed.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N11 HIGH ENERGY PIPE BREAK OUTSIDE CONTAINMENT (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: CU03 RWCU SYSTEM PIPE BREAK OUTSIDE CONTAINMENT (STEAM TUNNEL)

Major Discrepancies:

F-0255 WITH CU03 ACTIVE AND DELTA FLOW AT GREATER THAN 100 GPM, THE ISOLATION TIMER TIMES OUT. WHEN SYSTEM SHOULD ISOLATE, DELTA FLOW INDICATION SPIKES DOWN TO 0 GPM AND THEN RETURNS TO 100 GPM

F-0257 "STEAM TUNNEL LD AMB TEMP P632" ANNUNCIATOR WAS NOT RECEIVED UNTIL AMB TEMP WAS 155 DEGF AS READ ON RECORDER 1E31-R608 THIS ALARM SHOULD BE RECEIVED AT AN AMBIENT TEMPERATURE OF 135 DEGF IN STEAM TUNNEL.

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.06.ED06, LOSS OF 480V BUS
Revision Number: 03
ANSI/ANS 3.5 Section: 3.1.2 (03)(d) Plant Malfunctions; Loss of Distribution Bus

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Date Tested: 06/19/91

Run Time: 2.00 hours

Test Description: The purpose of this test is to verify proper response of simulator to the following 480v bus failures: EF1A, EF1B, EF1C, EF1D, F1A, F1B, F1C, F1D, F1E, F1F, F1G, XF1A. Initial conditions are as follows: 100% power, steady state. Failure is inserted to give respective bus power loss and alarms, recorders, indicating lights, isolations are verified to occur as associated plant procedure ONI-R23 1 or 2 states. Failure mode is such that the bus cannot be recovered by operator actions until trainer allows. Failures are verified to be inserted and deleted properly. At completion of tests, the plant will be at condition determined by severity of loads lost by respective bus as stated in ONI-R23 1 or 2.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-R23-1 LOSS OF AN ESSENTIAL 480V BUS (UNIT 1)
ONI-R23-2 LOSS OF A NON-ESSENTIAL 480 VOLT BUS (UNIT 1)

Malfunctions Tested: ED06 LOSS OF 480V BUS

Discrepancies: None

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.06.ED09, LOSS OF 125V DC BUS
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (03)(e) Plant Malfunctions; Loss of DC Instrument Bus

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Date Tested: 06/19/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify proper response of simulator to the following 125V DC bus failures: ED1A, ED1B, ED1C, D1A, and D1B. Initial conditions are 100% power, steady state. Failure is inserted to give respective bus power loss. For ED1A, ED1B, and ED1C, D1B alarms, recorders, indicating lights and for ED1B - RX SCRAM on Main Turbine trip from Reactor Core Isolation Cooling initiation signal are verified to occur as per ONI-R42 1,2, 3 or 5 respectively. For D1A annunciation loss except for "ANN PWR SUPPLY FAIL", recorders, indicating lights, L11 and L12 bus transfers to L10, Rx Recirc pumps tripped to off, are verified to occur as per ONI-R42-4. Failures are verified to be inserted and deleted properly. At completion of tests, the plant will be at 100% power with power restored to the buses, except for ED1B which will be recovering from a scram with power restored to ED1B bus; and for D1A, Reactor power will be lower due to loss of RX Recirc pumps, with power restored to D1A bus.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-R42-1 LOSS OF DC BUS ED-1-A (UNIT 1)
ONI-R42-2 LOSS OF DC BUS ED-1-B (UNIT 1)
ONI-R42-3 LOSS OF DC BUS ED-1-C (UNIT 1)
ONI-R42-4 LOSS OF DC BUS D-1-A (UNIT 1)
ONI-R42-5 LOSS OF DC BUS D-1-B (UNIT 1)

Malfunctions Tested: ED09 LOSS OF 125V DC BUS

Major Discrepancies:

F-0241 ON LOSS OF ED1B, DID NOT LOSE INDICATION OR CONTROL OF 1P45-C001B (EH1205). WE LATER FAILED ED1A AND EH1205 INDICATION WAS LOST - BELIEVE WRONG POWER SOURCE IS SIMULATED.

F-0245 ON LOSS OF ED1B, DID NOT LOSE CONTROL OR INDICATION OF (EF1D04). WE LATER FAILED ED1A AND EF1D04 INDICATION WAS LOST - BELIEVE WRONG POWER SOURCE IS SIMULATED.

F-0246 ON LOSS OF ED1B, DID NOT LOSE INDICATION OR CONTROL OF 1E12-C002B (EH1208). WE LATER FAILED ED1A06 AND EH1208 INDICATION WAS LOST - BELIEVE WRONG POWER SOURCE IS SIMULATED.

F-0247 ON LOSS OF EH11 - LOST INDICATION OF P45-C001B SINCE P45-C001B SHOULD BE POWERED FROM EH12, LOSS OF EH11 SHOULD HAVE NO EFFECT ON P45-C001B

F-0248 RCIC SYSTEM LOSES DIV 2 LOGIC POWER ON A LOSS OF ED1A (INDICATED ON OOS MATRIX AMBER LIGHTS) RCIC DIV 2 LOGIC/POWER LOSS SHOULD BE DRIVEN FROM ED1B, NOT ED1A.

Test T.4.5.1.06.ED09, LOSS OF 125V DC BUS

Major Discrepancies (continued):

F-0265 DID NOT RECEIVE POWER LOSS OOS INDICATORS FOR ESW B/ECC B WHEN LOSS OF ED1B WAS RUN. ED1B PROVIDES CONTROL POWER FOR BREAKERS, SHOULD RECEIVE OUT OF SERVICE LIGHTS AND ALARM.

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.06.ED17, LOSS OF 125V DC DISTRIBUTION PANEL
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 (03)(d) Plant Malfunctions; Loss of Distribution Bus

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Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the Simulator to a loss of 125V DC distribution panels. ED17 has seven cases (panels) in order to fail both safety-related and non-safety 125V DC distribution. The simulator is initialized to 100% power IC and individual cases of ED17 are inserted. Plant load lists are used to evaluate simulator performance and proper electrical distribution modeling. This includes checking for proper receipt of alarms, automatic actions, and loss of power to control circuits, components, and instrumentation. Each malfunction case is deleted to ensure proper restoration is achieved.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-R42-1 LOSS OF DC BUS ED-1-A (UNIT 1)
ONI-R42-2 LOSS OF DC BUS ED-1-B (UNIT 1)
ONI-R42-4 LOSS OF DC BUS D-1-A (UNIT 1)
ONI-R42-5 LOSS OF DC BUS D-1-B (UNIT 1)

Malfunctions Tested: ED17 LOSS OF 125V DC DISTRIBUTION PANEL

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.07.EG01, MAIN GENERATOR LOCKOUT RELAY TRIP
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (16) Plant Malfunctions; Generator Trip

Date Tested: 06/20/91

Run Time: 0.50 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a spurious trip of the Main Generator Lockout Relay, 86-1 and 86-2. The simulator is initialized to an IC at which it is operating at 100% steady state power. Malfunction EG01A is activated and operator action in accordance with ONI-N32 (Turbine and/or Generator Trip), and ONI-C71-1 (Reactor Scram) will be taken. All automatic actions associated with a generator trip from power are checked against those listed in the aforementioned ONI's. At the conclusion of the test, the plant will be stabilized in a post scram condition. EG01A is then deleted and it is verified that the 86-1 relay can be reset. EG01B is tested to the extent necessary to verify it can be inserted and deleted from the instructor station.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: EG01 MAIN GENERATOR LOCKOUT RELAY TRIP

Major Discrepancies: None

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.09.FW02, FEEDWATER SYSTEM PIPE BREAK INSIDE DRYWELL
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (20)(c) Plant Malfunctions; Feedwater Line Break, inside containment

Date Tested: 06/19/91

Run Time: 0.50 hour

Test Description: The purpose of this test is to verify the response of the simulator to a large feedwater line break within the drywell structure. This malfunction is a discrete variable with 100% severity representing a 20 inch double end pipe failure. The simulator is initialized to a 100% IC and the malfunction is inserted at 100% severity. Containment and Reactor Pressure Vessel response will be evaluated against Perry USAR Chapter 15.6.5. Automatic actions initiated in response to a large break LOCA are also verified to perform correctly. At the end of this test, the reactor is shutdown with High Pressure Core Spray and/or Reactor Core Isolation Cooling maintaining RPV level between Level 2 and Level 8. Reactor pressure will be slowly decreasing as a result of spray system injection. Drywell and Containment parameters degrading due to the loss of cooling systems. Lastly FW02 is verified as a non-recoverable type by attempting to delete it.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)

Malfunctions Tested: FW02 FEEDWATER SYSTEM PIPE BREAK INSIDE DRYWELL

Major Discrepancies:

F-0252 WITH MALF FW02 INSERTED AND N27-F200 OPEN & ALL FW PUMPS EXCEPT ONE RFBP SHUTDOWN, NOTED FEED FLOW PEGGED HIGH (~20 MLB/HR).

F-0253 AFTER FEEDWATER BREAK INTO THE DRYWELL WAS ISOLATED AND NCC RESTORED TO THE DRYWELL AHU'S, THE DRYWELL TEMPERATURES DID NOT DECREASE.

F-0270 INDICATED FW FLOW ON 1C34R607 PEGGED HIGH (>20 MLBM/HR) REGARDLESS OF THE AMOUNT OF "REAL" FW FLOW WHEN A FW BREAK IN DW IS ACTIVE.

F-0285 HPCS PUMP TRIPPED ON OVERCURRENT WHEN HPCS HI DRYWELL PRESSURE AUTO INITIATION SIGNAL WAS RECEIVED.

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.09.FW03, FEEDWATER SYSTEM PIPE BREAK OUTSIDE CONTAINMENT
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (20)(c) Plant Malfunctions; Feedwater Line Break, outside containment

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Date Tested: 06/20/91

Run Time: 1.75 hours

Test Description: The purpose of this test is to verify the response of the simulator to a large feedwater line break outside the Contain... t Structure within the Turbine Building Steam Tunnel. This malfunction is discrete, variable, with 100% severity representing a 20 inch double end pipe failure. The simulator is initialized to a 100% Power IC and the malfunction is inserted at 100% severity. The sequence of events and key parameters are checked against those in the Perry USAR Chapter 15.5.6. System response and isolations expected to occur on a break in the Steam Tunnel are also verified. The plant stabilizes in a post scram condition with the MSIV's closed. RPV level is maintained between level 2 and level 8 by the operation of High Pressure Core Spray and/or Reactor Core Isolation Cooling. RPV pressure is maintained initially by safety/relief valve operation and slowly decreases through the operation at spray systems. Lastly, FW03 is verified as a non-recoverable type by attempting to delete it.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-G33 REACTOR WATER CLEANUP SYSTEM (UNIT 1)

Malfunctions Tested: FW03 FEEDWATER SYSTEM PIPE BREAK OUTSIDE CONTAINMENT

Major Discrepancies:

F-0114 WITH THE MSIV'S ISOLATED DUE TO MF FW03 ACTIVE AT 10% (FW BREAK IN TUNNEL) MS LINE TEMPS OSCILLATED, THEN PEGGED HIGH WITHIN 1 MINUTE. (CAUSE SIM FAILURE ONE TIME, NOT THE NEXT).

F-0274 WITH A FW LINE BREAK IN THE STEAM TUNNEL, FW HDR FLOW AS SENSED BY 1C34N002B DID NOT INCREASE.

F-0275 WITH A FW PIPE BREAK IN THE STEAM TUNNEL, MALFUNCTION FLOW DROPPED TO 0 AFTER THE RFPT'S TRIPPED. RFBP'S, MFP WAS IN OPERATION WITH A FLOW PATH STILL AVAILABLE TO THE PIPE BREAK LOCATION. ENERGY INPUT OUT BREAK SHOULD CONTINUE UNTIL THE LEAK FLOW PATH IS ISOLATED.

F-0289 VERY LITTLE INCREASE IN STEAM TUNNEL AMBIENT TEMP WAS SEEN AND NO INCREASE WAS SEEN IN TB AMBIENT TEMP DURING A FW LINE BREAK IN STEAM TUNNEL. 400 DEG F FW FLASHING TO STEAM OUT THE BREAK SHOULD CAUSE A RAPID AND LARGE INCREASE IN STEAM TUNNEL/TB TEMPS.

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.12.IA01, AIR RECEIVER LEAK
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (02) Plant Malfunctions; Loss of Instrument Air

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Date Tested: 06/20/91

Run Time: 2.00 hours

Test Description: The purpose of this test is to verify the simulator's response separately to a loss of Service and Instrument Air System receivers. This malfunction contains 4 cases of which two, IA01A and IA01C will be evaluated for plant response. The other two, IA01B and D will be tested sufficiently to ensure they can be inserted and deleted only. The plant is initialized to a 100% IC and the malfunctions inserted at 100% severity which simulates a receiver end weld failure. Automatic actions and component failures are evaluated against those listed in ONI-P51/P52, loss of Service and/or Instrument Air. For IA01A the plant stabilizes still at power following the automatic closure of service air to instrument air system cross-connects. In the case of IA01B, the plant stabilizes in a post scram condition with the MSIV's closed and the instrument air system depressurized.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-P52 LOSS OF SERVICE AND/OR INSTRUMENT AIR

Malfunctions Tested: IA01 AIR RECEIVER LEAK

Major Discrepancies:

F-0277 ON A LOSS OF AIR, N23 SYSTEM DP DID NOT DECREASE TO APPROXIMATELY 0 PSID. N23 BYPASS VALVES 1N23F020 FAILS OPEN ON LOSS OF AIR, CAUSING SYSTEM DP TO DECREASE TOWARD 0 PSID.

F-0278 WITH A LOSS OF AIR EJECTORS DUE TO A LOSS OF INSTRUMENT AIR, MAIN CONDENSER VACUUM REMAINED CONSTANT. EXPECT TO SEE SOME AMOUNT OF VACUUM DEGRADATION WITH NO AIR EJECTORS IN OPERATION AND STEAM BEING DUMPED TO THE CONDENSER.

F-0279 DW VACUUM BKR MOV'S M16F010A AND F010B WERE CYCLING IN THE PRESENCE OF AN NS4 ISOLATION SIGNAL.

F-0280 WITH 80# AT FRONT STANDARD, THE MAIN TURBINE DID NOT TRIP.

F-0283 1C11F002A DID NOT FAIL CLOSED ON LOSS OF AIR.

F-0291 WHEN U1 SERVICE AIR RECEIVER RUPTURED, THE IA COMPRESSORS STARTED BUT UNLOADED AT 95#. COMPRESSOR START IN AUTO/ON-OFF MODE AND SHOULD NOT UNLOAD TILL ABOUT 101.5#.

F-0292 ON A LOSS OF INSTRUMENT AIR, THE AUX BOILER WAS STILL AVAILABLE.

Test T.4.5.1.12.IA01, AIR RECEIVER LEAK

Major Discrepancies (continued):

P-0294 ON LOSS OF AIR N26 VALVES DID NOT FAIL AS REQUIRED.

P-0252 AFTER MALF IA01 WAS INSERTED, THE FOLLOWING PROBLEMS WERE SEEN:

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.12.IA02, INSTRUMENT AIR LINE LEAK
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.10

Date Tested: 06/20/91

Run Time: 1.25 hours

Test Description: The purpose of this response is to verify the proper response of the Simulator to an instrument air system line leak. IA01 has five cases (A-E), to simulate line breaks in 5 separate areas of the plant. The malfunctions are variable (0-100%) where 100% equals a complete line severance (2 inch pipe). During the test, Cases A-C are fully tested for plant response. Cases D&F are verified to insert and delete from the instructor station. For each case tested, the simulator is initialized to a 100% power IC. IA02A(B,C is inserted at 100% severity. The proper response of the plant to the line break is checked using Perry off-normal instructions. Major effects checked include air operated components failing to the fail position air compressors starting at proper setpoints, and consequential effects of loss of air to various systems. Breaks occurring within the containment are verified to cause a pressure increase in the affected area. Due loss of air to the SCRAM air header, the reactor SCRAMS. At the end of the test, the plant is shutdown and stabilized in a post SCRAM condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-P52 LOSS OF SERVICE AND/OR INSTRUMENT AIR

Malfunctions Tested: IA02 INSTRUMENT AIR LINE LEAK

Major Discrepancies:

F-0271 WHEN AIR DISTRIB HDR PRESS IS LOW, AIR OPERATED VLVS GO TO FAIL POSITION BUT ONLY IN PROPORTION TO AIR HDR PRESS. THIS IS OK TO A POINT BUT RESPONSE IS UNREALISTIC IN RANGE OF .001 PSIG TO 1 PSIG.

F-0276 WITH A COMPLETE LOSS OF INSTRUMENT AIR TO LOADS OUTSIDE CONTAINMENT, THE CLEARWELL PUMPS DID NOT TRIP.

F-0281 ON A COMPLETE LOSS OF INSTRUMENT AIR TO LOADS OUTSIDE THE CONTAINMENT, 1N64F051C AND 1N64F051D DID NOT FAIL OPEN.

F-0286 WITH AN INSTRUMENT AIR LINE BREAK IN THE DRYWELL, DRYWELL PRESSURE DID NOT INCREASE.

F-0287 WITH AN AIR BREAK IN CONTAINMENT, CONTAINMENT PRESSURE DID NOT INCREASE.

F-0288 WITH A LOSS OF AIR TO THE DRYWELL, HEAT EXCHANGER COILS DID NOT AUTO SWAP TO THE B COILS FOR THE DRYWELL COOLERS.

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.15.MC01, CONDENSER AIR INLEAKAGE
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 (05)(a) Plant Malfunctions; Loss of Condenser Vacuum

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Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the simulator's response to a loss of condenser vacuum. Malfunction MC01 has 3 cases: MC01A - Main Condenser HP Shell, MC01B - Auxiliary condenser A, and MC01B - Auxiliary Condenser B. All three are variable in nature 0-100% with 100% equivalent to 300 scfm at 3 inHgA. MC01A is tested, B and C are verified to be inserted and deleted from the Instructor Station. The simulator is initialized to a 100% steady state IC with two circulating water pumps in service, and MC01A is inserted at 10% severity. Main and Auxiliary condenser vacuums are then verified to slowly degrade. Changes in plant Offgas system are also verified. The automatic actions outlined in ONI-N62 are verified to occur. The plant response is evaluated against that listed in the Perry USAR Chapter 15.2.5. The plant stabilizes in a post scram condition with MSIV's closed. RPV pressure is being maintained by use of safety/relief valves and RPV level control is on the Motor Feed Pump augmented as necessary by High Pressure Core Spray and/or Reactor Core Isolation Cooling.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N62 LOSS OF MAIN CONDENSER VACUUM (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: MC01 CONDENSER AIR INLEAKAGE

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.16.MS11, STEAM SEAL HEADER PRESSURE REGULATOR FAILURE
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.7.1

Date Tested: 06/20/91

Run Time: 0.60 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a failure of the Steam Seal Evaporator Header Pressure Control valve, 1N33-F070. The malfunction is variable 0-100% and both extremes are tested to view the affects. The simulator is initialized to a 100% IC with the SSE in service. The regulating valve is first failed full open and then full close. Indicating lights, alarms and metered instrumentation is then evaluated. Malfunction MS11 is then deleted and proper operation of the N33-F070 valve is observed. The plant remains stable at 100% power at the completion of this test.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-N33 STEAM SEAL SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-N62 LOSS OF MAIN CONDENSER VACUUM (UNIT 1)

Malfunctions Tested: MS11 STEAM SEAL HEADER PRESSURE REGULATOR FAILURE

Discrepancies: None

Evaluators: C. Persson, B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.17.NM01, SRM DETECTOR (PRE-AMP) FAILURE
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (21) Plant Malfunctions; Nuclear Instrumentation Failures

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Date Tested: 06/20/91

Run Time: 0.50 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a failure of Source Range Monitor Pre-amplifier. This malfunction has four cases: NM01A through D, are specific for each SRM. Each malfunction is variable C-100% output of the pre-amp. NM01A verifies upscale failures while NM01B, C, and D verify downscale failures. The simulator is initialized in a startup IC with power still in the source range. NM01A is inserted to 100% severity and the response is verified. SRM A counts increase to offscale, Reactor Period goes towards zero and remains positive until saturation of the pre-amp. Remote Function RP01 is used to remove the RPS shorting links to verify the scram function occurs. The plant stabilizes in a post scram condition and NM01A is deleted.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-C51(SRM) SOURCE RANGE MONITORING SYSTEM (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: NM01 SRM DETECTOR (PRE-AMP) FAILURE

Major Discrepancies:

F-0272 NM01 (FAIL TO 0%) WANTED TO ACTIVATE & IMMEDIATELY RAMP TO 100%; FAIL ACTIVATION (%) & SEVERITY/RAMP RATE DON'T MAKE ANY SENSE

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.17.NM02, IRM DETECTOR (PRE-AMP) FAILURE
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (21) Plant Malfunctions; Nuclear Instrumentation Failures

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Date Tested: 06/21/91

Run Time: 1.25 hours

Test Description: The purpose of this test is to verify proper response of the simulator to individual IRM failure upscale/downscale. Initial conditions are as follows: approximately 2% power, plant heatup in process. Failure is inserted to give IRM upscale (100% severity). Alarms, recorders, and indicating lights are verified to operate properly with 1/2 scram received. Failure is inserted to give IRM downscale (ramp to 0%). Alarms, recorders and indicating lights are verified to operate properly with rod block received. Failures are verified to be inserted and deleted properly. At completion of the tests, the plant is at 2%.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-C51(IRM) INTERMEDIATE RANGE MONITORING SYSTEM (UNIT 1,

Reference Type: Other
Best Estimate

Malfunctions Tested: NM02 IRM DETECTOR (PRE-AMP) FAILURE

Discrepancies: None

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.17.NM03, LPRM DETECTOR FAILURE
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (21) Plant Malfunctions; Nuclear Instrumentation Failures

Date Tested: 06/21/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify proper response of the simulator to individual LPRM failure upscale. Initial conditions are as follows: 75% power, all APRM's within 2% of each other with all LPRM signals to APRM's valid. Failure is inserted on one of the LPRM's to APRM "D" (severity level at 100%). Alarms, recorders and indicating lights are verified to operate properly with APRM "D" reaching approximately 77.5%. The failure is verified to be inserted and deleted properly. At completion of the test, the plant is at 75% power.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-C51(APRM) AVERAGE POWER RANGE MONITORING SYSTEM (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: NM03 LPRM DETECTOR FAILURE

Discrepancies: None

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.17.NM10, NEUTRON MONITORING DETECTOR DRIVE STUCK
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 (21) Plant Malfunctions; Nuclear Instrumentation Failures

Date Tested: 06/21/91

Run Time: 1.5 hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a stuck neutron detector (SRM/IRM). There are 12 cases of NM10 (A-L), one for each moveable detector. The failure is variable to allow for sticking the detector at any position between full in (0%) and full out (100%). The simulator is initialized to a shutdown IC in which the permissives for detector movement are satisfied. The malfunction is activated for various detectors at different severities and the detector is moved and verified to stick at the position corresponding to the selected severity (full in, full out, intermediate) by observing reactor period, detector counts, status lights, etc. Each case of NM10 is verified to activate and delete properly from the instructors console.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-C51(SRM) SOURCE RANGE MONITORING SYSTEM (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: NM10 NEUTRON MONITORING DETECTOR DRIVE STUCK

Discrepancies: None

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.18.OG03, OG SYSTEM LEAK UPSTREAM OF ADSORBERS
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.7.1

Date Tested: 06/23/91

Run Time: 0.25 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a break in the Offgas process piping upstream of the charcoal adsorbers. Malfunction OG03 is variable, 0-100%, and the break location is within the holdup pipe room. The simulation is initialized to a 100% steady state IC and the malfunction is inserted with 100% severity. Offgas parameters such as flow, component differential pressures and component temperatures are verified to decrease. Transport of radioactive material is verified by increases in holdup pipe room radiation levels and increases in the offgas vent pipe radiation levels. The plant remains at 100% steady state with offgas being processed to the offgas vent pipe via Offgas Building Ventilation System (M36). OG03 is then deleted to verify it can be cleared from the instructor station and a slow return to normal pre-test conditions is checked.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-D17 HIGH RADIATION LEVELS WITHIN PLANT (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: OG03 OG SYSTEM LEAK UPSTREAM OF ADSORBERS

Major Discrepancies:

F-0323 WHEN A D17 MONITOR EXCEEDS THE ALERT SETPOINT, THE HIGH ALARM LIGHT BLINKS, AND VICE VERSA. THIS WAS NOTED ON 2 OG RAD MONITORS. COULD BE A GENERIC PROBLEM.

F-0330 WITH MF OG03 ACTIVE, PARTICULATE AND IODINE RAD LEVELS DECREASED PROPORTIONAL TC GASEOUS ACTIVITY. PARTICULATE AND IODINE ACTIVITY CONTINUE TO INCREASE SINCE THEIR VALVES ARE CUMULATIVE (INTEGRAL) DUE TO FILTER IN SAMPLE SKID.

F-0332 OBSERVED NO INCREASE IN OG HOLD UP PIPE AREA RADIATION WHEN MALFUNCTION OG03 WAS ACTIVE AT 100% (LEAK IN HOLD UP PIPE).

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.18.OG04, OG SYSTEM LEAK DOWNSTREAM OF ADSORBERS
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.7.1

Date Tested: 06/23/91

Run Time: 0.25 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to an offgas process piping failure downstream of the adsorbers. Malfunction OG04 is variable, 0-100%, and the break location is at the inlet to the after filters upstream of N64-F061. The simulator is initialized to a 100% steady state IC and the malfunction is inserted with 100% severity. Offgas parameters such as individual train flow, indicated total system flow and component differential pressures are verified to change in the correct direction for the given break location. Offgas vent pipe radiation levels are verified not to change as this process has been treated by the adsorbers. The plant remains stable at 100% throughout this test. OG04 is deleted and the parameters return to their pretest values.

Baseline Data Used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-D17 HIGH RADIATION LEVELS WITHIN PLANT (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: OG04 OG SYSTEM LEAK DOWNSTREAM OF ADSORBERS

Discrepancies: None

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.21.RD03, CONTROL ROD DRIFT - IN
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 (12)(c) Plant Malfunctions; Drifting Control Rod

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Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a control rod drift caused by Directional Control Valve (DCV) EP-123 failure during rod motion. RD03 may be selected for any one or more of the 177 control rods. The plant is initialized to a 100% steady state IC and a partially withdrawn control rod is selected at the instructor station. With no rod motion, RD03 is verified to be passive. When rod motion is commanded by the operator, the selected rods's EP-123 fails to reclose following the completion of the insert (or withdrawal) sequence and an inward rod drift occurs. Changes in rod position and appropriate RC&IS alarms are verified to occur. Rod motion continues until RD03 is deleted from the instructor station. The plant remains steady at approximately 100% power after the test is completed.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
OM6: ARI PERRY UNIT ALARM RESPONSE INSTRUCTIONS

Reference Type: Other
Best Estimate

Malfunctions Tested: RD03 CONTROL ROD DRIFT - IN

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.21.RD04, CONTROL ROD DRIFT - OUT
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 (12)(c) Plant Malfunctions; Drifting Control Rod

Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a control rod drift outward caused by a stuck collet piston. RD04 may be selected for any one or more of the 177 control rods and the failure occurs following a command signal for rod motion. The simulator is initialized at a 100% steady state IC and a control rod that is not fully withdrawn is selected for failure at the instructor station. With no rod motion, the malfunction is verified to be passive. Following a withdraw sequence, the rod continues outward at a rate of about 1 inch per second. Changes in rod position and other RC&IS indications as well as a Rod Drift alarm are verified to occur. The rod drift can be stopped by the command of an insert signal or single rod scram using a remote function. The plant remains steady at approximately 100% power after the test is completed.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
OM6: ARI PERRY UNIT ALARM RESPONSE INSTRUCTIONS

Reference Type: Other
Best Estimate

Malfunctions Tested: RD04 CONTROL ROD DRIFT - OUT

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.22.RH02, RHR SYSTEM PIPE BREAK
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 (07) Plant Malfunctions; Loss of Shutdown Cooling

Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a break in a RHR Pump discharge pipe. RH02 has three cases: RH02A, B and C which simulate a break in each of the three RHR loops. The malfunction is variable, 0-100% with 100% equivalent to an 18 inch break. Since RHR has the ability to have up to three different suction sources, this test looks at all three separately. The simulator is initialized in a cold Shutdown state to test the fuel pool suction pa and a Hot Shutdown IC to test suppression pool cooling suction path and a in to test the shutdown cooling flowpath. In all three resets, the malfunction RH02 is tested to 100% severity. Flowrates, pressures, water inventory transfers and automatic system activations and isolations are verified to occur dependent on suction source and fluid temperatures. At the completion of each test, RH02 is deleted to ensure it can be cleared from the instructor station.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)

Malfunctions Tested: RH02 RHR SYSTEM PIPE BREAK

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.23.RP02, INADVERTENT INITIATION OF ARI
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (17) Plant Malfunctions; Failure in Reactivity Control System

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Date Tested: 06/22/91

Run Time: 0.50 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to an inadvertent initiation of Alternate Rod Insertion (ARI). RP02 has four cases A, B, C, and D, one for each channel of RRCS. All four cases will be inserted during this test. The simulator is initialized to a 100% steady state IC and each case is inserted individually. It is then verified that alarms, indications and automatic actions occur per system design resulting in a full core scram. After the ARI is completed, the malfunctions are deleted and following the appropriate time delay, it is verified that the ARI can be reset. The plant stabilizes in a post scram condition at the completion of the test.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-C71-1 REACTOR SCRAM (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: RP02 INADVERTENT INITIATION OF ARI

Discrepancies: None

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.25.SW01, NCC SYSTEM PROCESS PIPING LEAKAGE
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (08) Plant Malfunctions; Loss of Component Cooling

Date Tested: 06/22/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a loss of Nuclear Closed Cooling caused by a discharge header break. This malfunction is variable, 0-100% with 100% equivalent to a 30 inch pipe break. The simulator is initialized to a 100% steady state IC and SW01 is inserted with 100% severity. The response of NCC is verified to include a loss of system inventory, system flow and pressure. Components cooled by NCC are verified to heatup and the automatic actions listed in ONI-P43, Loss of Nuclear Closed Cooling are also verified. The plant stabilizes in a post scram condition with NCC cooled components continuing to degrade. Malfunction SW01 is then deleted and an increase in NCC surge tank level is verified.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-P43 LOSS OF NUCLEAR CLOSED COOLING (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: SW01 NCC SYSTEM PROCESS PIPING LEAKAGE

Major Discrepancies:

F-0303 WHEN P47 A CHILLER TRIPPED (P904) THE CHILLER TRIP ANNUNCIATOR WAS NOT RECEIVED.

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.25.SW02, SERVICE WATER SYSTEM PROCESS PIPING LEAKAGE
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (06) Plant Malfunctions; Loss of Service Water

Date Tested: 06/22/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a loss of Plant Service Water caused by an underground pipe break. This malfunction is variable, 0-100% with 100% equivalent to a 54 inch pipe break. The simulator is initialized to a 100% steady state IC and SW02 is inserted with 100% severity. The response of Service Water is verified to include a loss of system flow and discharge pressure. Components cooled by Service Water are verified to heatup and the automatic actions listed in ONI-P41, Loss of Service Water, are also verified. The plant stabilizes in a post scram condition and the loads serviced by Service Water continue to degrade. Malfunction SW01 is then deleted.

Baseline Data used F : Reference

Reference Type: Off-Normal Operating Procedures
ONI-P41 LOSS OF SERVICE WATER (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: SW02 SERVICE WATER SYSTEM PROCESS PIPING LEAKAGE

Major Discrepancies:

F-0304 BEFORE FAILURE TBCC AND MAIN LUBE OIL TEMP CONTROLLERS ARE ALREADY FULLY OPEN WITH LAKE TEMP AT 55 DEG. THEY ARE NORMALLY AT SOME THROTTLED POSITION, ESP. AT 55 DEG LAKE TEMP. NEED ABILITY TO CONTROL MLO AND TBCC TEMPS "IN BAND" UP TO LAKE TEMP OF 80 DEG F.

F-0306 WITH SW02 INSERTED COOLING TWR MAKE UP VLV HAD TO CLOSE DOWN BECAUSE IT STILL HAD FLOW. FLOW TO CT BASIN WENT UP WHEN WEIR LEVEL WENT TO 0.

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.27.TH01, RECIRC LOOP RUPTURE (DBA LOCA)
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 (01)(b) Plant Malfunctions; LOCA inside containment

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Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a DBA Recirc Loop Rupture (LOCA). This malfunction has two cases, one for each recirc loop. Each simulates a catastrophic failure of the recirc piping between the RPV and the pump suction valve. The simulator is initialized to a 100% steady state IC and TH01A is inserted. The response of the RPV, Containment and Drywell are evaluated against the Perry USAR Chapter 15.6.5. Following core reflood to above TAF, all ECCS except LPCS is terminated to verify a floodable volume exists such that RPV water level does not decrease to less than 2/3 core height. Peak cladding temps are verified not to exceed 2200 deg F during this transient. The Containment Isolation function and ECCS initiation functions are verified to occur. The plant stabilizes in a post scram condition. TH01A is then verified to not delete until the simulator is reset. TH01B is inserted after reinitialization and verified to also be non-recoverable.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Malfunctions Tested: TH01 RECIRC LOOP RUPTURE (DBA LOCA)

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.27.TH19, RPV LEVEL INST REFERENCE LEG BREAK
Revision Number: 01
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.6.2

Date Tested: 06/23/91

Run Time: 1.00 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to a rupture of a reactor vessel level instrument sensing line (reference leg). This malfunction has several cases which allow the failure of different lines, either inside or outside the drywell. The simulator is initialized at 100% Power IC. Individually, each generic case is activated and the response of the simulator is checked. For each case, proper indication for a small break LOCA are observed. The overall plant response is judged depending on the location of the break (inside/outside the Drywell). The response of attached level and pressure sensing instruments is also verified to be proper. The malfunction is non-recoverable, so the simulator must be reset to test each case. At the end of each test run, the reactor is in a Post-Scram condition, with ECCS systems injecting to the RPV to maintain water level.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Malfunctions Tested: TH19 RPV LEVEL INST REFERENCE LEG BREAK

Major Discrepancies:

F-0298 TRANSMITTERS FOR CERTAIN VESSEL PRESSURE/LEVEL INSTRUMENTS ARE NOT ASSIGNED TO ANY INDICATOR "LEG" OR ARE ASSIGNED TO WRONG REF/VARIABLE LEG.

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.27.TH20, RPV LEVEL INST VARIABLE LEG BREAK
Revision Number: later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.6.2

Date Tested: Not Complete

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a rupture of a reactor vessel level instrument sensing line (reference leg). This malfunction has several cases which allow the failure of different lines, either inside or outside the drywell. The simulator is initialized at 100% Power IC. Individually, each generic case is activated and the response of the simulator is checked. For each case, proper indication for a small break LOCA are observed. The overall plant response is judged depending on the location of the break (inside/outside the Drywell). The response of attached level and pressure sensing instruments is also verified to be proper. The malfunction is non-recoverable, so the simulator must be reset to test each case. At the end of each test run, the reactor is in a Post-Scram condition, with ECCS systems injecting to the RPV to maintain water level.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Malfunctions Tested: TH20 RPV LEVEL INST VARIABLE LEG BREAK

Major Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.27.TH21, POWER/FLOW INSTABILITIES (IEB 88-07 Supplement 1)
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.1 (07) Normal Plant Evolutions; Operations with Less than Full Coolant Flow

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Date Tested: 06/23/91

Run Time: 0.40 hour

Test Description: The purpose of this test is to verify the proper response of the simulator to neutron flux instabilities that may result from operation in the region of the BWR-6 power to flow map known as the "red zone" or high power/low flow region. While the exact magnitude of the instabilities is currently unknown for the Perry Plant, and intentional operation in this region is prohibited (which accounts for lack of comparison data), some data is available from the LaSalle Plant event, and predictions of possible response have been made by General Electric. This malfunction is available to allow training of Perry Operators on the possible effects of this phenomena. The effects can be varied from minor oscillations to severe oscillations which result in a Scram. The test is initiated from the 100% Power operating condition. The malfunction is inserted at 100% severity. No effects are seen since initial operation is occurring outside the prohibited region of the power to flow map. Flow is then intentionally lowered to enter the red zone. Out of phase neutron flux oscillations are verified to occur on individual LPRM's. Oscillations are also verified on the APRM's, resulting in a Reactor Scram due to high flux. At the end of the test, the plant is stabilized in a post-Scram condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-B33-1 REACTOR RECIRCULATION FLOW CONTROL MALFUNCTION (UNIT 1)
ONI-C51 UNEXPLAINED CHANGE IN REACTOR POWER OR REACTIVITY (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: TH21 POWER/FLOW INSTABILITIES (IEB 88-07 Supplement 1)

Major Discrepancies:

F-032? WHEN MALFUNCTION TH21 WAS INSERTED WHILE OPERATING IN THE REGION OF INSTABILITY OF THE POWER/FLOW MAP, DID NOT RECEIVE LPRM/APRM UPSCALE/DOWNSCALE ALARMS IN AN OSCILLATING FASHION. WHEN OSCILLATIONS OCCUR THEY SHOULD BE RANDOM UPSCALE AND DOWNSCALE IN DIFFERENT REGIONS OF THE CORE AS FLOW OSCILLATIONS OCCUR IN DIFFERENT REGIONS OF THE CORE.

P-0966 WHEN AT 100% POWER ACTIVATION OF THIS MALF CAUSES A SCRAM IMMEDIATELY.

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.27.TH27, MAIN STEAM LINE RUPTURE IN STEAM TUNNEL
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (20)(b) Plant Malfunctions; Main Steam Line Break, outside containment

Date Tested: 06/23/91

Run Time: 0.50 hour

Test Description: The purpose of this test is to verify the Simulator response to a Main Steam Line (MSL) Rupture in the Steam Tunnel. The break location is downstream the MSIV's and isolable. The malfunction is single case, nonrecoverable. The test is initiated from 100% Power operating condition. The failure is activated and plant response is verified against Perry USAR chapter 15.6.4 analysis. Proper response of alarms, automatic actions, and key parameter trends are checked. At the end of the test, the plant is stabilized in a Post-Scram condition with the containment and Reactor Vessel isolated, pressure controlled by safety/relief valves, and RPV level being maintained by the feedwater system, augmented by the High Pressure Core Spray and Reactor Core Isolation Cooling systems. Failure TH27 is verified to be non-recoverable by attempting to delete it from the Instructors Station.

Baseline Data used For Reference

Reference Type: Other
Best Estimate

Malfunctions Tested: TH27 MAIN STEAM LINE RUPTURE IN STEAM TUNNEL

Major Discrepancies:

F-0327 NO RADIA[illegible] [illegible]FECTS OBSERVED DURING STEAM LEAK IN STEAM TUNNEL. EXPECT IN[illegible] [illegible]ED RAD LEVELS FOR TB/HB VENT EXHUAST, AUX BLDG EXHAUST, TB [illegible]REA RAD MONITORS.

F-0328 ALL WESTRONICS MULTIPOINT TEMP RECORDERS ARE DISPLAYING TEMPS THAT ARE MORE THAN 10% IN ERROR ON THE HIGH END. SCALE NORMALIZATION IS LINEAR, METER SCALES ARE NOT.

Evaluators: B. Panfil, H. DeBoer, G. Minshall

Test T.4.5.1.28.TU01, MAIN SHAFT OIL PUMP DEGRADATION
Revision Number: 02
ANSI/ANS 3.5 Section: 3.1.2 (15) Plant Malfunctions; Turbine Trip

Date Tested: 06/23/91

Run Time: 0.50 hour

Test Description: The purpose of this test is to verify the proper response of the Simulator to a Turbine Trip caused by a failure of the Main Shaft Oil Pump (MSOP). The malfunction is single case, variable severity where 100% severity represents complete pump failure (0 PSI discharge head). The test is initiated from the 100% Power operating condition. The malfunction is activated on a 2 minute ramp from 0 to 100% severity. During the pump degradation, oil pressures in the Main Turbine Lube Oil system (MTLO) are monitored. Automatic actions are verified as pressure lowers such as standby and emergency pumps starting. A Main Turbine trip on low MSOP pressure is observed to occur, resulting in a plant shutdown. At the end of the test, the plant is stabilized in the Post-Scram condition with the motor suction pump, turning gear oil pump, and all bearing lift pumps running.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)

Reference Type: Other
Best Estimate

Malfunctions Tested: TU01 MAIN SHAFT OIL PUMP DEGRADATION

Discrepancies: None

Evaluators: B. Panfil, H. DeBoer, G. Minshall

ATP TEST SECTION T.4.5.3

MALFUNCTION SCENARIOS

Overview of testing performed in this section

This series of Tests were performed to verify the proper response of the Simulator to the set of Malfunction Scenarios for which the simulator is being certified. These malfunction scenarios are comprised of one or more System Level and/or Component Level Failures, combined with I/O overrides, Event Triggers, and/or Remote Functions. Each Malfunction Scenario can be used singly or in combination during the course of an Operator Training or Evaluation session. Prior tests in the Acceptance Test Procedure (ATP) tested the individual Failures which are contained in a scenario, so the primary purpose of the section 4.5.3 tests were to verify the proper overall plant response as predicted for each malfunction scenario. The test abstracts include a description of which specific combination of failures (including rates and severities), overrides, triggers, and remote functions were tested. The tests are divided into two basic categories: those which validate malfunctions required for simulation of types of Perry USAR accidents (those which result in observable indications on control room instrumentation and for which the simulator has been determined to be applicable), and those which support the current performance based operator training curricula and were not tested in previous ATP sections T.4.5.1 or T.4.4.2.

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.5.3.01, LOSS OF FEEDWATER HEATING (RRFC IN AUTO)	3	(N/C)*
T.4.5.3.02, LOSS OF FEEDWATER HEATING (RRFC IN MANUAL)	4	(N/C)*
T.4.5.3.03, FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND	5	(N/C)*
T.4.5.3.04, PRESSURE REGULATOR FAILURE-OPEN	6	(N/C)*
T.4.5.3.05, INADVERTENT SAFETY/RELIEF VALVE OPENING	7	(N/C)*
T.4.5.3.06, INADVERTENT RHR SHUTDOWN COOLING OPERATION	8	(N/C)*
T.4.5.3.07, PRESSURE REGULATOR FAILURE-CLOSED	9	(N/C)*
T.4.5.3.08, GENERATOR LOAD REJECT WITH BYPASS VALVES	10	(N/C)*
T.4.5.3.09, GENERATOR LOAD REJECT WITHOUT BYPASS VALVES	11	(N/C)*
T.4.5.3.10, TURBINE TRIP	12	(N/C)*
T.4.5.3.11, LOSS OF AC POWER (LOSS OF AUX TRANSFORMER)	13	(N/C)*
T.4.5.3.12, FAILURE OF RESIDUAL HEAT REMOVAL SHUTDOWN COOLING	14	(N/C)*
T.4.5.3.13, RECIRC FLOW CONTROL FAILURE-DECREASING (BOTH FCV'S)	15	(N/C)*
T.4.5.3.14, RECIRCULATION PUMP SEIZURE	16	(N/C)*
T.4.5.3.15, ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP	17	(N/C)*
T.4.5.3.16, RECIRCULATION PUMP SHAFT SHEAR	18	(N/C)*
T.4.5.3.17, RECIRC FLOW CONTROL FAILURE-INCREASING (BOTH FCV'S)	19	(N/C)*
T.4.5.3.18, CONTROL ROD DROP ACCIDENT	20	(N/C)*

Tests included in this Section:	Page:	No. of Open SDR's:
T.4.5.3.19, INADVERTENT HIGH PRESSURE CORE SPRAY STARTUP	21	(N/C)*
T.4.5.3.20, ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)	22	(N/C)*
T.4.5.3.21, LOSS OF OFF-SITE POWER	23	(N/C)*
T.4.5.3.22, PEI MALFUNCTION SCENARIO #1	24	(N/C)*
T.4.5.3.23, PEI MALFUNCTION SCENARIO #2	26	(N/C)*
T.4.5.3.24, PEI MALFUNCTION SCENARIO #3	28	(N/C)*
T.4.5.3.25, PEI MALFUNCTION SCENARIO #4	30	(N/C)*
T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)	32	(N/C)*
T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1	33	(N/C)*
T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2	35	(N/C)*
T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3	37	(N/C)*
T.4.5.3.30, EVALUATION MALFUNCTION SCENARIO #4	39	(N/C)*
T.4.5.3.31, EVALUATION MALFUNCTION SCENARIO #5	40	(N/C)*
T.4.5.3.32, EVALUATION MALFUNCTION SCENARIO #6	42	(N/C)*
T.4.5.3.33, EVALUATION MALFUNCTION SCENARIO #7	44	(N/C)*
T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8	46	(N/C)*
T.4.5.3.35, EVALUATION MALFUNCTION SCENARIO #9	48	(N/C)*
T.4.5.3.36, EVALUATION MALFUNCTION SCENARIO #10	50	(N/C)*
T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11	52	(N/C)*
T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12	54	(N/C)*
T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13	56	(N/C)*
T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14	58	(N/C)*
T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15	60	(N/C)*
T.4.5.3.42, PEI MALFUNCTION SCENARIO #5 (PART 2)	62	(N/C)*

ANSI/ANS 3.5 Reference: 3.1.2 Plant Malfunctions

* This test has not been completed (N/C) or has been performed but has one or more unresolved major Simulator Discrepancy Reports associated with it. Please see the individual test abstract for these discrepancies which constitute exceptions to ANSI/ANS 3.5 section 3.1.2.

Test T.4.5.3.01, LOSS OF FEEDWATER HEATING (REACTOR RECIRC FLOW CONTROL IN AUTO)
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.1.1

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a partial loss of Feedwater Heating accident with Reactor Recirc Flow Control in AUTO as described in the Perry USAR, section 15.1.1. The test is initialized at 100% Power operation. Extraction Steam to the number 5A FW heater is isolated by inserting a component failure on the supply MOV. The sequence of events and key parameter response is checked as per the USAR description. At the end of the test, conditions are as follows: Core flow is about 80% of initial, Reactor Power is 106% of initial, feedwater injection temperature has lowered, Turbine load and steam flow are nearly the same as initial.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B10 PREPARATION FOR FINAL FEEDWATER TEMPERATURE REDUCTION OPERATION

Reference Type: Off-Normal Operating Procedures
ONI-C51 UNEXPLAINED CHANGE IN REACTOR POWER OR REACTIVITY (UNIT 1)
ONI-N36 LOSS OF FEEDWATER HEATING (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.1.1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.1-1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.1-1 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: MV03 MOV Spurious valve closure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.02, LOSS OF FEEDWATER HEATING (REACTOR RECIRC FLOW CONTROL IN MANUAL)
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.1.1

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a partial loss of Feedwater Heating accident with Reactor Recirc Flow Control in Manual as described in the Perry USAR, section 15.1.1. The test is initialized at 89% Power operation. Extraction Steam to the number 6A FW heater is isolated by inserting a component failure on the supply MOV. The sequence of events and key parameter response is checked as per the USAR description and the results of Startup Test STI-N27-023B. At the end of the test, conditions are as follows: Core flow is the same as initial, Reactor Power has increased, but a SCRAM has not occurred, feedwater injection temperature delta is less than or equal to 100 deg F. The acceptance criteria for this test is identical to the STI acceptance criteria for the plant startup test.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
FTI-B10 PREPARATION FOR FINAL FEEDWATER TEMPERATURE REDUCTION OPERATION

Reference Type: Off-Normal Operating Procedures
ONI-C51 UNEXPLAINED CHANGE IN REACTOR POWER OR REACTIVITY (UNIT 1)
ONI-N36 LOSS OF FEEDWATER HEATING (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.1.1 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: MV03 MOV Spurious valve closure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.03, FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.1.2

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Feedwater Controller Failure accident (maximum demand) as described in the Perry USAR, section 15.1.2. The test is initialized at 100% power. The malfunction is initiated by inserting component level failure CN01 for the feedwater Master Level Controller 1C34R0600 at 100% severity. This causes both operating Reactor Feed Pumps to ramp to the high speed stop (5450 RPM), injecting the maximum amount of feedwater. The reactor scrams and the feed pumps automatically trip on high level (219.5 inches above TAF), mitigating the transient. The sequence of events and trend of key parameters is checked per the USAR description. At the end of the test, the simulated plant is stabilized in the Post-Scram condition with reactor level being controlled by the HPCS and RCIC systems.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-C51 UNEXPLAINED CHANGE IN REACTOR POWER OR REACTIVITY (UNIT 1)
ONI-N36 LOSS OF FEEDWATER HEATING (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.1.2 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.1-3 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.1-3 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: CN01 Controller Auto/manual failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.04, PRESSURE REGULATOR FAILURE-OPEN
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.1.3

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Reactor Vessel Pressure Regulator Failure (open) accident as described in the Perry USAR, section 15.1.3. The test is initialized at 100% Power, EOC life. To simulate the initial conditions of the plant as described in the USAR analysis, the MSIV stroke time is set to maximum allowable (5 seconds) with System Level Failure MS03. The Pressure regulator is placed in TEST mode (simulating that one channel is out of service) with remote function TC08. This prevents an automatic transfer to the non-failed channel. The malfunction is initiated by failing the main steam line pressure transmitter input to the inservice regulator with component level failure PT01 at 100% severity or failed upscale. The pressure regulator responds by rapidly opening all the valves under its control: the Turbine Control Valves, and the Steam Bypass Valves. The rapid rise in steam flow causes a RPV depressurization and vessel level swell, initiating a Reactor SCRAM due to high level. The sequence of events and response of key parameters are verified per the USAR description. At the end of the test, the reactor is shutdown, the Main feedwater pumps and the Main Turbine are tripped, the MSIV's are isolated, reactor pressure is being controlled by the Safety and Relief Valves, and reactor level is being controlled by the HPCS and RCIC systems.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-C85-2 PRESSURE REGULATOR FAILURE - OPEN (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.1.3 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.1-4 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.1-4 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: MS03 MSIV CLOSURE TIME VARIANCE
PT01 Process Transmitter Variable failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.05, INADVERTENT SRV OPENING
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.1.4

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to an inadvertent SRV (Safety and Relief Valve) opening as described in the Perry USAR, section 15.1.4. The test is initialized at the 100% Power Operating Condition. One SRV control switch is failed in the open position (simulating a short in the switch) by use of I/O override. Plant parameters are verified to respond as described in the USAR analysis, and as backed up by Perry Startup Test Data from the performance of SVI-B21-T2005. At the completion of the test, Reactor Power remains at 100%, and RPV pressure is slightly lowered from initial.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.1.4 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.1-5 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.06, INADVERTENT RHR SHUTDOWN COOLING OPERATION
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.1.6

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a improper initiation of Shutdown Cooling flow as described in the Perry USAR, section 15.1.6. The test is initialized with the plant in a cold startup, IC08. At the point of reactor criticality (low in the source range), actions are taken to initiate one RHR Loop in the shutdown cooling mode. At the reference plant, this would require operator error, as there are no plant malfunctions which could result in this condition. Plant parameters are verified to respond as per the USAR description. At the end of the test, the reactor is verified to have scrammed on high neutron flux.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
IOI-1 COLD STARTUP
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.1.6 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.07, PRESSURE REGULATOR FAILURE-CLOSED
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.1

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Reactor Vessel Pressure Regulator Failure (closed) accident as described in the Perry USAR, section 15.2.1. The test is initialized at 100% Power, EOC life. The C85 Pressure regulator is placed in TEST mode (simulating that one channel is out of service) with remote function TC08. This prevents an automatic transfer to the non-failed channel. The malfunction is initiated by failing the main steam line pressure transmitter input to the inservice regulator with component level failure PT01 (variable severity transmitter output failure) at 0% severity or failed downscale. The pressure regulator responds by rapidly closing all the valves under its control: the Turbine Control Valves, and the Steam Bypass Valves. The rapid reduction in steam flow causes a RPV pressurization and neutron flux excursion. A Reactor SCRAM occurs due to high flux. The sequence of events and response of key parameters are verified per the USAR description. At the end of the test, the reactor is shutdown, the Main feedwater pumps and the Main Turbine are tripped, and reactor pressure is being controlled by the Safety and Relief Valves.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-C85-1 PRESSURE REGULATOR FAILURE - CLOSED (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.2.1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.2-1 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: PT01 Process Transmitter Variable failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.08, GENERATOR LOAD REJECT WITH BYPASS VALVES
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.2

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Generator Load Reject (Bypass available) as described in the Perry USAR, section 15.2.2. The test is initialized at the 100% Power operating condition, IC19. The load reject is initiated by opening the Generator Output Breakers. The main turbine is verified to trip due to Power/Load unbalance. The plant response is verified per the USAR description in that a SCRAM and Recirc Pump Trip is initiated due to the turbine trip. At the end of the test, the reactor plant is shutdown, and pressure is being maintained by the bypass valves.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)
ONI-S11 LOSS OF OFF-SITE POWER

Reference Type: Plant Data - Analyses
USAR 15.2.2 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-2 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.2-2 PNPP UPDATED SAFETY ANALYSIS REPORT

Reference Type: Plant Data - Startup Test Results
STI-B21-027/2 DATA REQUEST RESPONSE - TURB TRIP & GEN LD REJ

Malfunctions Tested: CB01 Spurious breaker trip

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.09, GENERATOR LOAD REJECT WITHOUT BYPASS VALVES
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.2

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Generator Load Reject (Bypass not available) as described in the Perry USAR, section 15.2.2. The test is initialized at the 100% Power operating condition, IC19. System level failure TC04 is activated at 0% severity for all Turbine Bypass Valves to prevent them from opening. The load reject is then initiated by opening the Generator Output Breakers. The main turbine is verified to trip due to Power/Load unbalance. The plant response is verified per the USAR description in that a SCRAM and Recirc Pump Trip is initiated due to the turbine trip. The peak pressure observed is verified to not exceed the USAR predicted maximum. At the end of the test, the reactor plant is shutdown, and pressure is being maintained by the SRV's.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)
ONI-S11 LOSS OF OFF-SITE POWER

Reference Type: Plant Data - Analyses
USAR 15.2.2 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-3 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.2-3 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: CB01 Spurious breaker trip
TC04 BYPASS VALVE FAILURE

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.10, TURBINE TRIP
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.3

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Turbine Trip as described in the Perry USAR, section 15.2.3. Three turbine trip scenarios are performed by this test. The first and second are initiated from the 100% Power Operating condition, EOC. The third is initiated from 30% Power. The first turbine trip is initiated with bypass valves available. The second and third trips are initiated after first failing the Turbine Bypass valves closed with failure TC04 at 0% severity. For each turbine trip, the sequence of events is verified per the USAR description. The response of key parameters is verified by comparison to the USAR figures and tables. At the end of each trip, the Simulator is stable in the Post-SCRAM condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.2.3 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-4 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-5 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.2-4 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.2-5 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.11, LOSS OF AC POWER (LOSS OF AUX TRANSFORMER)
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.6

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Loss of Normal AC Power as described in the Perry USAR, section 15.2.6. The test is initialized at the 100% Power operating condition, IC19. The transient is initiated by inserting Failure TF01 on Aux Transformer 110-PY-B, resulting in a loss of electrical power to 13.8KV busses L12 and L12, and all non-1E (BOP) 4160V and 480V AC busses. The sequence of events and response of key parameters is verified to occur as per the USAR description, with allowance for assumed worst case conditions in the USAR analysis. At the end of the test the Reactor is shutdown and isolated, normal feedwater has tripped off, reactor pressure is being controlled by SRV's, and RPV level is being maintained by the HPCS and RCIC systems.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-R22-1 LOSS OF AN ESSENTIAL AND/OR A STUB 4.16KV BUS (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.2.6 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-8 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.2-12 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: TF01 Loss of Transformer

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.12, FAILURE OF RHR SHUTDOWN COOLING
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.9

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Loss of Shutdown Cooling (RHR) following a loss of off-site power as described in the Perry USAR, section 15.2.9. The test is initialized at the 100% power condition, IC19. Initial conditions are established to represent the USAR analysis conditions: Suppression Pool level of 18.0 ft, Suppression Pool Temperature of 90 deg F, and lake temperature of 80 deg F. One emergency diesel generator is taken out of service. Valve 1E12-F008 is failed with MOV component failure MV06 (single passive failure). The transient is initiated by tripping all off-site grid tie breakers, resulting in a loss of off-site power. The immediate effects are verified per the USAR description and ONI-S11, Loss of Off-Site Power. The plant stabilizes in a post-scram condition with the RPV isolated, pressure being maintained by SRV's and level being maintained with the high pressure injection system HPCS and RCIC. After about 30 minutes, actions are taken to commence a forced fast cooldown using SRV's. After 12 minutes, the Reactor Pressure has been lowered to 100 psig. Due to failure of normal shutdown cooling, an alternate (emergency) method is established per the USAR description, following directions given in ONI-E12-2, Loss of Shutdown Cooling using LPCI injection and flowing out through open SRV's. The ability to maintain long-term fuel cooling and reduce reactor pressure below 100 psig in this lineup is verified. At the end of the test, plant conditions are as described above.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-E12-2 LOSS OF SHUTDOWN COOLING (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.2.9 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: MV06 MOV Fail as is (mechanical binding)

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.13, RECIRC FLOW CONTROL FAILURE-DECREASING (BOTH FCV'S)
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.3.2

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Reactor Recirculation Flow Control Valve Closure as described in the Perry USAR, section 15.3.2. The test is initialized at 100% Power operating condition, IC19. The RRC flow control valves controllers are failed to 0% output with Controller Failure CN01. The flow control valves are verified to stroke closed at their maximum rate. When the flow control valves are at their minimum position, total core flow has been reduced to about 25% of rated. The rapid decrease in power and flow results in a level excursion causing a SCRAM on high reactor level. The simulator response is verified per the USAR description. At the end of the test the plant is stabilized in the Post-SCRAM condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-B33-1 REACTOR RECIRCULATION FLOW CONTROL MALFUNCTION (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.3.2 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.2-4 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.3-4 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: CN01 Controller Auto/manual failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.14, RECIRCULATION PUMP SEIZURE
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.3.3

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Recirculation Pump Shaft Seizure as described in the Perry USAR, section 15.3.3. The test is initialized at 100% Power, IC19. The transient is initiated by inserting Pump Component failure CP02 on Reactor Recirc Pump A. The sudden loss of flow in the A Loop results in a vessel level excursion, causing a Reactor SCRAM due to high level. The response of key parameters is verified per the USAR description. At the end of the test, the plant is stabilized in the post-SCRAM condition.

Baseline Data used For Reference

Reference Type: Plant Data - Analyses
USAR 15.3.3 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.3-5 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.3-5 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: CP02 Pump Shaft seizes

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.15, ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.4.4

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to an improper Recirculation Loop restoration as described in the Perry USAR, section 15.4.4. The test is initialized at 100% Power, IC19. To prepare for the transient, Reactor Recirc Pump A is tripped, the A Recirc Loop is isolated and allowed to cool down to 100 deg F. The temperature transmitter which normally provides an interlock to prevent restoring a cold, idle loop to service is failed to read normal (hot) temperature with transmitter failure PT01. The idle loop is then unisolated, and Recirc Pump A is started normally per the system operating instructions (with exception of verifying proper delta temperatures), resulting in a reactivity excursion. Key parameter response is verified per the USAR description (the reactor does not SCRAM). Final plant conditions are similar to initial.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-B33 REACTOR RECIRCULATION SYSTEM

Reference Type: Plant Data - Analyses
USAR 15.4.4 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.4-1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.4-1 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: PT01 Process Transmitter Variable failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.16, RECIRCULATION PUMP SHAFT SHEAR
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.3.4

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Recirculation Pump Shaft Shear as described in the Perry USAR, section 15.3.4. The test is initialized at 100% Power, IC19. The transient is initiated by inserting Pump Component failure CP01 on Reactor Recirc Pump A. The sudden loss of flow in the A Loop results in a vessel level excursion, causing a Reactor SCRAM due to high level. The response of key parameters is verified per the USAR description. At the end of the test, the plant is stabilized in the post-SCRAM condition.

Baseline Data used For Reference

Reference Type: Plant Data - Analyses
USAR 15.3.4 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: CP01 Pump Shaft breaks

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.17, RECIRC FLOW CONTROL FAILURE-INCREASING (BOTH FCV'S)
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.4.5

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Reactor Recirculation Flow Control Valve Opening as described in the Perry USAR, section 15.4.5. The test is initialized at 100% Power operating condition, IC19. To approximate the initial conditions of the Analysis, reactor power is reduced to about 54% by reducing Reactor Core flow to 33% of rated. The RRC flow control valves Master Controller is failed to 100% output with Controller Failure CN01. The flow control valves are verified to stroke open at their maximum rate. Prior to reaching their full open position, a reactor SCRAM occurs due to the power transient. The simulator response is verified per the USAR description. At the end of the test the plant is stabilized in the Post-SCRAM condition.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-B33-2 LOSS OF ONE OR BOTH RECIRCULATION PUMPS (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.4-5 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15.4-5 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.4-5 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: CN01 Controller Auto/manual failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.18, CONTROL ROD DROP ACCIDENT
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.4.9

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Control Rod Drop accident described in the Perry USAR, section 15.4.9. The test is initialized with a reactor startup in progress, and the reactor critical. Control Rods are adjusted to achieve worst case conditions per the USAR analysis (50% rod density). Two failures are activated on a high worth rod; RD01 to stick the rod full in, and RD02 to uncouple the rod from its drive mechanism. The Rod Pattern Controller is bypassed and the selected CRDM is fully withdrawn (rod stays full in). Failure RD01 is deleted, resulting in the rod travelling at a rapid speed from full in to full out. The resulting neutron flux transient is verified to initiate a reactor SCRAM. The final conditions approximate the initial conditions with the exception of all Control Rods being inserted due to SCRAM.

Baseline Data used For Reference

Reference Type: Plant Data - Analyses
USAR 15.4.9 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.4-9 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: RD01 STUCK CONTROL ROD
RD02 UNCOUPLED CONTROL ROD

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.19, INADVERTENT HIGH PRESSURE CORE SPRAY STARTUP
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.5.1

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to an iradvertent High Pressure Core Spray system initiation as described in the Perry USAR, section 15.5.1. The test is initialized at 100% Power, IC19. The transient is initiated by failing the HPCS manual initiation switch with I/O override (simulates short in switch). The response of key plant parameters is verified to respond as per the USAR description. The final plant conditions are similar to the initial with reactor power slightly higher and reactor pressure slightly lower.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-E12-1 INADVERTENT INITIATION OF ECCS/RCIC (UNIT 1)

Reference Type: Plant Data - Analyses
USAR 15.5-1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15.5-1 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.20, ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.8

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Failure of the Reactor Protection System to initiate and complete a SCRAM during a major plant transient (vessel isolation) as described in the Perry USAR, section 15.8. Initial conditions: The simulator is initialized at 100% Power, EOC. The ATWS malfunction is activated at 100% severity (passive failure); 100% severity will result in no rod motion. A transient is then induced by inserting a spurious MSIV closure (vessel isolation) as described in the USAR. Critical parameters are recorded and compared to USAR analytical data. Automatic actions and alarms are verified. Two minutes after the MSIV closure, the SLC (boron injection) pumps are started. Final Conditions: The reactor is shutting down due to boron injection, the control rods are full out, the RPV is isolated. SRV's are controlling Rx Pressure, augmented by RCIC operation. As the core shuts down, reactor vessel level is restored by HPCS and RCIC injection.

Baseline Data used For Reference

Reference Type: Plant Data - Analyses
USAR 15.8 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR APPENDIX 15C PNPP UPDATED SAFETY ANALYSIS REPORT
USAR FIGURE 15C-1 PNPP UPDATED SAFETY ANALYSIS REPORT
USAR TABLE 15C-3 PNPP UPDATED SAFETY ANALYSIS REPORT

Malfunctions Tested: RD15 ATWS

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.21, LOSS OF OFF-SITE POWER
Revision Number: Later
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions, USAR Accident 15.2.6

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify the proper response of the simulator to a Loss of Off-Site Power as described in the Perry USAR, section 15.2.6. Initial Conditions: The simulator is initialized to 18% steady state power IC with the main turbine on-line. To facilitate comparison to plant Startup Test data, reactor power is increased to 23%. The transient is initiated by simultaneously tripping the main turbine and opening the Off-site Supply breakers L1003 and L2003. Pertinent parameters are recorded and analyzed for comparison to Perry Unit 1 startup test data. The acceptance criteria for this test is the same criteria used for the startup test STI-R43-031. Final Conditions: the reactor is shutdown and isolated, RCIC and HPCS are maintaining RPV water level greater RPV Level 2, RCIC operation is controlling RPV pressure <1033 psig (SRV operation is allowed by the test criteria). Parameters are sufficiently stable that operators may take action to restore power to the Plant.

Baseline Data used For Reference

Reference Type: Off-Normal Operating Procedures
ONI-S11 LOSS OF OFF-SITE POWER

Reference Type: Plant Data - Analyses
USAR 15.2.6 PNPP UPDATED SAFETY ANALYSIS REPORT

Reference Type: Plant Data - Startup Test Results
STI-R43-031/2 DATA REQUEST RESPONSE - LOSS OF TURBINE GEN AND OFFSITE POWER

Malfunctions Tested: None

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.22, PEI MALFUNCTION SCENARIO #1
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting PEI (Plant Emergency Instructions) Malfunction Scenario #1. This scenario is used during the training of Perry Plant Licensed Operator candidates. The scenario guide consists of 3 parts, each designed to be run in 50 minutes. Each part is tested separately. Each part is begun with the simulator initialized at the 100% (EOC) power operating condition. Part 1 starts with RPS division 1 tripped to meet a Technical Specification LCO. APRM B is failed upscale, but a scram fails to occur. Test operators take action per PEI-B13 to initiate and complete a reactor shutdown using ARI (alternate rod insertion). At the end of Part 1, the plant is shutdown. Part 2 begins with a loss of main condenser vacuum, forcing a rapid plant shutdown. Vacuum continues to degrade and initiates a Reactor Vessel isolation, forcing entry into PEI-B13. A cycling SRV complicates both RPV pressure and level control, and is mitigated by removing the SRV control power fuses. The ability to exercise procedures PEI-E12 and PEI-G42 is validated when they are used to control Suppression Pool Level and Temperature. Part 3 begins by failing all Feedwater flow to the RPV. The loss is caused by a failed level switch tripping all Reactor Feedwater Booster pumps, which in turn trips all running normal Feedwater Pump Turbines. The Reactor scrams on low level and forces entry into PEI-B13. Reactor Level restoration is attempted by the use of HPCS and RCIC systems. These systems are failed, resulting in a complete loss of normal and emergency feed. A rapid reactor depressurization is performed to allow injection by low pressure ECCS. At the end of each part, the Reactor is shutdown with all Control Rods inserted with plant conditions being controlled as directed by the Plant Emergency Instructions.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)

Reference Type: Other
PEI Malfunction Scenario OT-3034-01A

Test T.4.5.3.22, PEI MALFUNCT.ON SCENARIO #1

Malfunctions Tested:
AD01 C ING SRV
CB01 Spurious breaker trip
CB06 Breaker fails in current position (loss of control power)
CN01 Controller Auto/manual failure
CP02 Pump Shaft seizes
ED07 LOSS OF 120V BUS
MC01 CONDENSER AIR INLEAKAGE
NM04 APRM OUTPUT FAILURE
RD17 LOSS OF CRD PUMP LUBE OIL
RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting operator training in accordance with PEI Malfunction Scenario #2. This scenario consists of three 50 minute exercises (parts). Each part is tested separately. At the beginning of each exercise, the simulator is initialized at the 100% Power operating condition, EOC. Part 1 starts with a spurious RRCS feedwater runback which results in a Reactor Scram due to low water level. While controlling reactor parameters per PEI-B13, a spurious MSIV isolation forces operators to change the mode of pressure control. Part 2 starts by failing 2 SRV's open, forcing entry into PEI-G42 and PEI-E12 to control Suppression Pool Temperature and Level. A fast reactor shutdown is performed when pool temperature limits are exceeded. A passive failure of the Reactor Protection System to initiate a SCRAM occurs, and alternate rod insertion (ARI) is used to insert control rods and complete the reactor shutdown. A failure of all high pressure injection systems occurs, resulting in the operators rapidly lowering reactor pressure to allow low pressure ECCS injection. Part 3 begins by failing the Control Rod Drive (CRD) pump and causing CRDM HCU accumulators to discharge, forcing operators to perform a reactor shutdown. Failures in the normal RPS system and the Alternate Rod Insertion system result in an ability to shutdown the reactor with control rods. Actions are performed in accordance with PEI-B13 to reduce Reactor core flow and RPV level to lower power. A CRD pump is recovered and control rods are inserted by the normal drive method. In each of the above exercises, the test ends with the reactor shutdown and plant parameters being controlled as directed by the plant normal and emergency instructions.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
IOI-8 SHUTDOWN BY MANUAL REACTOR SCRAM
SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
SOI-C71 RPS POWER SUPPLY DISTRIBUTION (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)

Test T.4.5.3.23, PEI MALFUNCTION SCENARIO #2

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)

Reference Type: Other
PEI Malfunction Scenario OT-3034-02A

Malfunctions Tested:
- CB01 Spurious breaker trip
- CB03 Breaker Auto trip logic failure
- CB06 Breaker fails in current position (loss of control power)
- MV03 MOV Spurious valve closure
- RD05 CONTROL ROD ACCUMULATOR FAULT
- RD15 ATWS
- RD17 LOSS OF CRD PUMP LUBE OIL
- RP01 EPA TRIP
- RP03 FAILURE OF ARI TO INITIATE
- RP04 INADVERTENT RRCS FW RUNBACK, RX RECIRC DOWNSHIFT, LOW FREQ MG TRIP
- RV03 Relief Valve Fails open
- RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

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Date: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Licensed Operator Training in accordance with PEI Malfunction Scenario #3. The scenario is divided into two parts. In each, the simulator is initialized at power, 50% power for the first, and 95% power near end of life for the second. The sequence of events for the first part of the scenario is: while operating at 50% power, two SRV's inadvertently open on their safety settings and remain stuck open. Operator actions are performed per PEI-E12 and PEI-G42 to control Suppression Pool temperature. Rising pool temperature necessitates a fast reactor shutdown. Control rod insertion fails by RPS and ARI. The Rx is shutdown by injecting boron with SLC (Standby Liquid) and normal rod insertion. Actions are also taken to reduce power by limiting injection and lowering Reactor Level. The sequence for the second part is: While operating at 95% power and inadvertent MSIV isolation occurs and the reactor is shutdown when ARI is initiated on high RPV pressure. Seven control rods fail to insert but are later inserted using CRDH. The loss of all high pressure injection systems result in RPV level decreasing to top of active fuel. At this point, the reactor is depressurized. Low pressure ECCS systems inject to restore reactor water level. The plant is then aligned for normal shutdown cooling. At the end of each part of the test, the reactor is shutdown using boron in case one and rods in case two.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
IOI-8 SHUTDOWN BY MANUAL REACTOR SCRAM
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E21 LOW PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)

Reference Type: Other
PEI Malfunction Scenario OT-3034-03A

Test T.4.5.3.24, PEI MALFUNCTION SCENARIO #3

Malfunctions Tested:
CB01 Spurious breaker trip
CB07 Breaker fails to close
MV06 MOV Fail as is (mechanical binding)
PT01 Process Transmitter Variable failure
RD01 STUCK CONTROL ROD
RD15 ATWS
RP03 FAILURE OF ARI TO INITIATE
RV03 Relief Valve Fails open
RY01 Relay Fails de-energized
RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.25, PEI MALFUNCTION SCENARIO #4
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Licensed Operator Training in accordance with PEI Malfunction Scenario #4. The scenario is divided into three parts. In each, the simulator is initialized at 100% power near end of core life. The sequence of events for the first part of the scenario: A small break LOCA (malfunction TH02A at 1% severity) results in increasing Drywell Pressure. ECCS initiation occurs at 1.68 psig sensed drywell pressure. The leak severity is increased to cause worsening containment conditions. Actions are performed in accordance with PEI-B13 to mitigate the high containment pressures. The sequence of events for the second part of the scenario: A small break in MSL between MSIV's occurs while operating at power. A failure of the MSL guard pipe coupled with a failure of the inboard MSIV to close results in a high energy release into the containment. When drywell pressure exceed 1.68 psig, PEI-B13 and PEI-D23 are entered. The Reactor is shutdown, and systems restored to remove heat from the containment structure. The reactor is depressurized and containment sprays are initiated to control containment pressure. The sequence of events for the third part of the scenario: A loss of drywell cooling is caused by a series of equipment failures associated with the drywell cooling system. The reactor is shutdown while drywell parameters are controlled using PEI-D23-2 and PEI-D23-3. Final plant conditions: In all three above scenarios, the reactor is shutdown and conditions are improving due to actions taken in response to PEI direction.

Baseline Data used For Reference

Reference Type: Emergency Cperating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL
PEI-D23-1 CONTAINMENT TEMPERATURE CONTROL
PEI-D23-2 DRYWELL & CONTAINMENT PRESSURE CONTROL

Reference Type: Normal Operating Procedures
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-P51/52 SERVICE & INSTRUMENT AIR SYSTEM

Reference Type: Other
PEI Malfunction Scenario OT-3034-04A

Test T.4.5.3.25, PEI MALFUNCTION SCENARIO #4

Malfunctions Tested:	AV03 Air Operated Valve Fails as is CB06 Breaker fails in current position (loss of control power) CP03 Pump Head loss (flow degradation) HX02 Heat Exchanger Tube leak PC01 INCREASED DW/CNTMT BYPASS LEAKAGE TC04 BYPASS VALVE FAILURE TH02 RECIRC LOOP PIPING BREAK TH28 MAIN STEAM LINE BREAK INSIDE GUARD PIPE
Discrepancies:	Unknown
Evaluators:	Later

Test T.4.5.3.26, PEI MALFUNCTION SCENARIO #5 (PART 1)
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

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Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Licensed Operator Training in accordance with PEI Malfunction Scenario #5 (part 1). Initial Plant Conditions: the simulator is initialized at 100% power near end of core life. Sequence of events: During severe weather a loss of feedwater and station blackout occur which results in a reactor shutdown. PEI-B13 actions are taken and steam cooling is conducted. PEI-M51/56 is entered in order to mitigate the effects of hydrogen generation. The Division 1 Diesel Generator is restored following fuel oil system repairs and RPV water level is restored promptly. The containment is vented until actual hydrogen concentrations can be determined.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL
PEI-M51/56 HYDROGEN CONTROL

Reference Type: Normal Operating Procedures
SOI-C71 RPS POWER SUPPLY DISTRIBUTION (UNIT 1)

Reference Type: Other
PEI Malfunction Scenario OT-3034-05A

Malfunctions Tested:
CB01 Spurious breaker trip
CB06 Breaker fails in current position (loss of control power)
CN01 Controller Auto/manual failure
DG03 DIESEL GEN SPEED GOVERNOR FAILS
DG06 FUEL OIL DAY TANK LEAK
ED05 LOSS OF 4.16 KV BUS
RC04 RCIC GOVERNOR VALVE FAILURE
RD15 ATWS
RP01 EPA TRIP
RP03 FAILURE OF ARI TO INITIATE
RY02 Relay Fails as is
SL05 SLC INJECTION PIPING LEAK
TF01 Loss of Transformer
TH15 GROSS FUEL FAILURE

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to validate the Simulator response during an operator training/evaluation exercise scenario. The initial test conditions are: Unit operating at full power at EOL conditions (coastdown), with the B Steam Air Ejector, and the B Control Rod drive pump unavailable. A surveillance instruction (SVI-C11-T1003) is performed which exercises control rods. A single rod is uncoupled (system level failure RD02) and then stuck (system level failure RD01). The rod is inserted and disarmed. An inadvertent HPCS Initiation signal is generated (active failure of bistable component 1B21N0667C/G). The HPCS Diesel Generator is failed when the start signal is received (passive failure DG03, case C). A failure of the A Steam Air Ejector is simulated (active failure of AOV components 1N62F0140A and 1N62F0170A) resulting in a loss of Condenser Vacuum. The Main Turbine trips on low vacuum. Normal operator actions are performed following a Reactor SCRAM per Perry Operating Procedures. Condenser vacuum continues to degrade, resulting in a full vessel isolation. The RCIC system is started to restore RPV level and control RPV pressure following the Main Steam Isolation. At the end of the test, the reactor is shutdown and isolated, the turbine has been shutdown, the RCIC system is running to maintain RPV level. This test allows for verifying the response of the simulator to a multiple failure scenario which may be used during operator examination evaluations. The ability to mitigate the effects of the various failures using Perry Plant Operating Procedures is also validated.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
SOI-B33 REACTOR RECIRCULATION SYSTEM

Reference Type: Off-Normal Operating Procedures
ONI-E12-1 INADVERTENT INITIATION OF ECCS/RCIC (UNIT 1)

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-01A

Test T.4.5.3.27, EVALUATION MALFUNCTION SCENARIO #1

Malfunctions Tested:
AV02 Air Operated Valve Fails closed
BS02 Bistable Spurious trip
CB05 Breaker fails in current position (mechanical seizure)
DG03 DIESEL GEN SPEED GOVERNOR FAILS
MC01 CONDENSER AIR INLEAKAGE
RD01 STUCK CONTROL ROD
RD02 UNCOUPLED CONTROL ROD

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.1.28, EVALUATION MALFUNCTION SCENARIO #2
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation scenarios used during Initial License Operator training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator actions are required to properly verify simulator response. The simulator is initialized to 96% reactor power with a division 3 outage in progress and minor BOP equipment out of service. RHR loop B is in the Suppression Pool Cooling mode of operation. APRM channel A fails downscale and a half scram is inserted to meet the requirements of Technical Specifications. A sequential loss of condensate and feedwater pumps occurs causing a Reactor Scram on low reactor water level. The Motor Feed pump is initially used for level control, but trips. The Reactor Core Isolation Cooling System (RCIC) fails to automatically start, but a manual start is successful. RPV pressure is maintained by the Pressure control system and RPV level is restored prior to reaching RPV Level 1 and level is brought back to the normal operating band of 185" to 215" using the Plant Emergency Instructions. At the end of the scenario, the Reactor scram procedure is being used to restore plant conditions.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
IOI-3 POWER CHANGES
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-N21 CONDENSATE SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-N27 FEEDWATER PUMP TRIP (UNIT 1)

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-02A

Test T.4.5.3.28, EVALUATION MALFUNCTION SCENARIO #2

Malfunctions Tested:	BS01	Bistable Fails to trip
	CB01	Spurious breaker trip
	CB06	Breaker fails in current position (loss of control power)
	FW08	FEEDWATER PUMP LOSS OF LUBRICATING OIL
	FW08	FEEDWATER PUMP LOSS OF LUBRICATING OIL
	NM04	APRM OUTPUT FAILURE
Discrepancies:	Unknown	
Evaluators:	Later	

Test T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License Operator training and Licensed Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator actions are required to properly evaluate simulator response. The simulator is initialized to 80% reactor power with one of the RFPT's out of commission. Other minor BOP equipment is out of service as well as RHR loop B. The "A" pressure regulator fails low, resulting in a transfer to the backup regulator. The Motor Feed Pump experiences a signal failure requiring entry into Off-Normal Instruction ONI-C34. A leak develops in division 1 diesel generator day tank requiring that the diesel be declared inoperable and placed in secured status. A Main Circ Water tube rupture occurs and a Fast Reactor Shutdown is performed. The Reactor Protection System fails to shutdown the reactor, however Alternate Rod Insertion is successful in inserting all control rods. The Reactor Core Cooling Isolation System is started for RPV level control and Plant Emergency Instruction PEI-B13 is exited to the Reactor Scram procedure. Actions are taken to isolate the feedwater and condensate systems in accordance with ONI-N61.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
IOI-3 POWER CHANGES
SOI-C34 FEEDWATER CONTROL SYSTEM (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-N27 FEEDWATER SYSTEM (UNIT 1)
SOI-R43 DIVISION 1 & 2 DIESEL GENERATOR SYSTEM (UNIT 1)
SOI-R45 DIVISION 1 & 2 DIESEL GENERATOR FUEL OIL SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-N61 CONDENSER TUBE LEAK/ORGANIC INTRUSION (UNIT 1)

Reference Type: Surveillance Procedures
SVI-R10-T5217 ELECTRICAL DISTRIBUTION SYSTEM ENERGIZATION CHECK

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-03A

Test T.4.5.3.29, EVALUATION MALFUNCTION SCENARIO #3

Malfunctions Tested:
CB06 Breaker fails in current position (loss of control power)
DG06 FUEL OIL DAY TANK LEAK
FW04 FEED PUMP LOGIC FAILURE
MC02 CONDENSER TUBE LEAK
PT01 Process Transmitter Variable failure
RY02 Relay Fails as is
SW07 LOSS OF COMPONENT COOLING - TBCC

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.30, EVALUATION MALFUNCTION SCENARIO #4
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License Operator training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. The simulator is initialized to 100% power with minor BOP equipment out of service. A power reduction is performed to conduct a Turbine Valve Exercise Test. Control Rod accumulator problems are encountered, followed by a trip of the operating CRD Pump. The pump is successfully restarted. The Hot Surge Tank level control valve fails causing a high level and isolation of heater 4, as well as Heater 6A with subsequent entry into ONI-N36. A sequential loss of Stator Water Cooling occurs causing a turbine load set runback and a Reactor Scram on high reactor pressure. PEI-B13 is used to control RPV level and pressure. During the pressure increase SRV's open as required, but one SRV fails to reclose. Actions are taken to close the valve by removing the appropriate fuses. The plant is stabilized and actions addressed by ONI-C71-1 (Reactor Scram) are being used at the end of the scenario.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-C71-1 REACTOR SCRAM (UNIT 1)

Reference Type: Surveillance Procedures
SVI-N31-T1151 MAIN TURBINE VALVE EXERCISE TEST

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES - A

Malfunctions Tested:
BS02 Bistable Spurious trip
CB01 Spurious breaker trip
CB06 Breaker fails in current position (loss of control power)
CN02 Controller Auto failure
MV02 MOV Spurious valve opening
MV03 MOV Spurious valve closure
RD05 CONTROL ROD ACCUMULATOR FAULT

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.31, EVALUATION MALFUNCTION SCENARIO #5
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 90% power, EOC conditions. The Motor Feed Pump is out of service for repair. The suppression pool is at an elevated temperature due to leaking SRVs. ESW and ECC loops "A" are in operation to support the anticipated startup of RHR loop "A" in the suppression pool cooling mode of operation. The sequence of events for this test are as follows; RHR loop "A" is started up in the suppression pool cooling mode. A shaft seizure occurs on Circ Water Pump A. Condenser vacuum degrades to the point of a Turbine Load Set Runback and a Recirc Flow Control Valve Runback. After these runbacks are reset, reactor power is increased to the maximum allowed by the existing condenser vacuum. A spurious Recirc Pump trip occurs and actions are taken to recover the tripped Recirc pump. The TBCC suction line pipe ruptures, causing a complete loss of TBCC. The temperatures of cooled components rapidly increase, and a Fast reactor Shutdown is performed. Vacuum further degrades and the MSIVs automatically close. PEI B13 is entered to stabilize the plant. RPV level is maintained by operation of RCIC and HPCS, however the HPCS Pump trips on overcurrent shortly after starting. RPV pressure is maintained by operation of RCIC and intermittent SRV operation. Final plant conditions are as follows; The reactor is shutdown and isolated with RPV level and pressure under control.

Baseline Data used For Reference

Reference Type: Normal Operating Procedures
IOI-3 POWER CHANGES
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-N21 CONDENSATE SYSTEM (UNIT 1)
SOI-N27 FEEDWATER SYSTEM (UNIT 1)
SOI-N32/39 MAIN TURBINE & TURNING GEAR SYSTEM (UNIT 1)
SOI-N64/62 OFF-GAS/CONDENSER AIR REMOVAL SYSTEM
SOI-P42 EMERGENCY CLOSED COOLING SYSTEM (UNIT 1)
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures

ONI-B33-2 LOSS OF ONE OR BOTH RECIRCULATION PUMPS (UNIT 1)
ONI-P44 LOSS OF TURBINE BUILDING CLOSED COOLING (UNIT 1)

Reference Type: Surveillance Procedures
SVI-B33-T1168 IDLE RECIRCULATION LOOP TEMPERATURE & FLOW

Test T.4.5.3.31, EVALUATION MALFUNCTION SCENARIO #5

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-05A

Malfunctions Tested:
CB01 Spurious breaker trip
CB06 Breaker fails in current position (loss of control power)
CP02 Pump Shaft seizes
MC01 CONDENSER AIR INLEAKAGE
MV01 MOV Fail as is (loss of control power)

SW03 TBCC SYSTEM PROCESS PIPING LEAKAGE

Discrepancies: Unknown

Evalu[illegible] Later

Test T.4.5.3.32, EVALUATION MALFUNCTION SCENARIO #6
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

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Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 95% power, EOC conditions. One Automatic Depressurization System (ADS) valve is inoperable with its fuses removed and the Division 2 diesel generator is operating unloaded. During the test an MSIV stroke test is performed and the diesel is loaded to bus EH12. A fire in an ECCS room cooling panel occurs and the room cooling fans controlled from that panel become inoperable. A plant shutdown is started. An inadvertent division 2 initiation signal occurs, causing, among other things, nuclear closed cooling (NCC) to be isolated to the containment and drywell. Temperatures rise to all the cooled components in these areas, eventually forcing the equipment to be manually or automatically secured. Some ECCS equipment fails to start on the initiation signal. Plant Emergency Instructions are entered to restore NCC to the containment and drywell. Final plant conditions are the plant shutdown with level and pressure being controlled in the normal band in accordance with PEI-[illegible] containment and drywell parameters improving due to actions taken in accordance with PEI-D23-2 and PEI-D23-3.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
- PEI-D23-2 DRYWELL & CONTAINMENT PRESSURE CONTROL
- PEI-D23-3 DRYWELL TEMPERATURE CONTROL

Reference Type: Normal Operating Procedures
- SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
- SOI-G41(FPCC) FUEL POOL COOLING & CLEANUP SYSTEM (UNIT 1)
- SOI-M15 ANNULUS EXHAUST GAS TREATMENT SYSTEM (UNIT 1)
- SOI-M25/26 CONTROL ROOM HVAC & EMERGENCY RECIRCULATION SYSTEM
- SOI-P43 NUCLEAR CLOSED COOLING SYSTEM
- SOI-R43 DIVISION 1 & 2 DIESEL GENERATOR SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
- ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)
- ONI-P43 LOSS OF NUCLEAR CLOSED COOLING (UNIT 1)

Test T.4.5.3.32, EVALUATION MALFUNCTION SCENARIO #6

Reference Type: Surveillance Procedures
SVI-B21-T2001 MSIV FULL STROKE OPERABILITY TEST

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-06A

Malfunctions Tested: CB05 Breaker fails in current position (mechanical seizure)
RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.33, EVALUATION MALFUNCTION SCENARIO #7
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

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Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 96% power, EOC conditions, with minor BOP equipment out of service for repair. The sequence of events for the test are as follows; A periodic test instruction is performed of the reactor feed pump turbines. A failure of the reactor recirculation flow control valve position device causes a FCV lockup. RFPT B trips due to a failure of its lube oil system, and power is automatically run back by the operable RR FCV. Recovery of the failed RR FCV loop is accomplished and power is increased to level allowed by the feedwater pumps available (80%). An SRV opens inadvertently and actions taken in accordance with ONI-B21-1 close the valve but it fails to fully reseat. PEI G42 and PEI E12 are used to cool the suppression pool and restore suppression pool level. Suppression pool temperature approaches 110 deg F and a Fast Reactor Shutdown is performed. The reactor scram procedure is entered and while the plant is being stabilized, a fire breaks out in the control room. The control room is evacuated and ONI C61 actions are completed. Final plant conditions are the reactor shutdown with level and pressure being controlled in accordance with ONI C71-1, one SRV leaking and the suppression pool slowly heating up.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-E12 SUPPRESSION POOL TEMPERATURE CONTROL

Reference Type: Normal Operating Procedures
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-N21 CONDENSATE SYSTEM (UNIT 1)
SOI-P43 NUCLEAR CLOSED COOLING SYSTEM

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)
ONI-B33-1 REACTOR RECIRCULATION FLOW CONTROL MALFUNCTION (UNIT 1)
ONI-C71-1 REACTOR SCRAM (UNIT 1)

Test T.4.5.3.33, EVALUATION MALFUNCTION SCENARIO #7

Reference Type: Periodic Test Procedures
PTI-N27-P0003 REACTOR FEED PUMP TURBINE STANDBY OIL PUMP OPERATION

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-07A

Malfunctions Tested:
CB01 Spurious breaker trip
CB06 Breaker fails in current position (loss of control power)
PT01 Process Transmitter Variable failure
RV02 Relief Valve Stuck
RV03 Relief Valve Fails open
TH14 RECIRC FCV HYDRAULIC POWER UNIT OIL HI TEMP

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operatcr Training

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Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 96% power, EOC conditions, with minor BOP equipment out of service for repair, as well as the High Pressure Core Spray System (HPCS) out of service. The sequence of events for this test are as follows: A periodic test instruction is performed on the Main Lube Oil system. A failure of the RCIC exhaust diaphragm pressure transmitters causes an inadvertent RCIC isolation to occur. A loss of an electrical bus causes RFPT A to trip, and shortly thereafter, a loss of all feedwater occurs due to a loss of all RFBPs. Plant Emergency Instruction PEI B13 is used in an attempt to restore water level, and when the last HP injection system is lost (CRDH A), the reactor is depressurized, first using the turbine bypass valves, then when level decreases to less than top of active fuel, an emergency depressurization is performed. The RPV Level 1 LOCA signal causes the containment and drywell parameters to degrade and actions are taken to restore these parameters using the containment, drywell, and suppression pool PEIs. Final plant conditions are the reactor shutdown and depressurized with RPV level restored to 185" to 215" using low pressure injection systems. Conditions in the drywell, containment, and suppression pool are improving.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL
PEI-E12 SUPPRESSION POOL TEMPERATURE CONTROL

Reference Type: Normal Operating Procedures
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-C51(IRM) INTERMEDIATE RANGE MONITORING SYSTEM (UNIT 1)
SOI-C51(SRM) SOURCE RANGE MONITORING SYSTEM (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-M51/56 COMBUSTIBLE GAS CONTROL SYSTEM & HYDROGEN IGNITERS (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-R23-2 LOSS OF A NON-ESSENTIAL 480 VOLT BUS (UNIT 1)

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-01E

Test T.4.5.3.34, EVALUATION MALFUNCTION SCENARIO #8

Malfunctions Tested: BS02 Bistable Spurious trip
CB01 Spurious breaker trip
CB04 Breaker Auto close logic failure
ED16 LOSS OF 480V MOTOR CONTROL CENTER (MCC)
R[illegible]7 LOSS OF CRD PUMP LUBE OIL

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.35, EVALUATION MALFUNCTION SCENARIO #9
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator actions are required to properly evaluate simulator response. The simulator is initialized to 96% power, EOC conditions. The Motor Feed Pump, as well as other minor BOP components are out of commission for repair. An operating Hotwell Pump strainer clogs, requiring a shift of HW pumps. An SRV inadvertently opens, causing entry into ONI-B21-1. Actions taken in this Off Normal Instruction are successful in closing the SRV (fuses to the SRV solenoid are removed). A failure of the manual initiation feature associated with the RCIC system cause the Reactor Core Isolation Cooling System to initiate. This initiation causes a trip of the Main Turbine and the Reactor Feed Pump Turbines. The reactor scrams and level and low reactor level 3 is quickly reached, causing entry into PEI-B13. As the plant conditions are being stabilized, a complete loss of level indication occurs. RPV flood is performed to ensure adequate core cooling. Final plant conditions are the plant shutdown with all control rods inserted, with adequate core cooling assured by maintaining MRF pressure with available low pressure and high pressure injection systems with the reactor depressurized.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-C51(IRM) INTERMEDIATE RANGE MONITORING SYSTEM (UNIT 1)
SOI-C51(SRM) SOURCE RANGE MONITORING SYSTEM (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-M51/56 COMBUSTIBLE GAS CONTROL SYSTEM & HYDROGEN IGNITERS (UNIT 1)
SOI-N21 CONDENSATE SYSTEM (UNIT 1)
SOI-N27 FEEDWATER SYSTEM (UNIT 1)
SOI-P42 EMERGENCY CLOSED COOLING SYSTEM (UNIT 1)
SOI-P43 NUCLEAR CLOSED COOLING SYSTEM
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)

Test T.4.5.3.35, EVALUATION MALFUNCTION SCENARIO #9

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)
ONI-C71-1 REACTOR SCRAM (UNIT 1)

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-02E

Malfunctions Tested: BS01 Bistable Fails to trip
BS02 Bistable Spurious trip
CP03 Pump Head loss (flow degradation)
PT01 Process Transmitter Variable failure

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.36, EVALUATION MALFUNCTION SCENARIO #10
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and Licensed Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator actions are required to properly evaluate simulator response. The simulator is initialized to 38% reactor power, ready to shift Reactor Recirculation Pumps to fast speed in order to continue power ascension. A division 3 outage is in progress and minor BOP equipment is out of service for repair. APRM E fails upscale causing a half scram. The APRM channel is bypassed and the half scram is reset. Problems are encountered with main condenser vacuum and a power reduction is required. As vacuum continues to degrade, a full scram signal is generated but the reactor protection system fails to shut down the reactor. Actions are taken to initiate Alternate Rod Insertion but the reactor stays at power. Further actions are taken in accordance with PEI-B13, including SLC injection and manual rod insertion. A stuck open SRV further degrades plant conditions. As power is reduced to less than 4% by the actions previously mentioned, ARI is finally successful in inserting all control rods. Final plant conditions include actions in progress to cool the suppression pool.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
IOI-3 POWER CHANGES
IOI-8 SHUTDOWN BY MANUAL REACTOR SCRAM
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E22B DIVISION 3 DIESEL GENERATOR (UNIT 1)
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)
ONI-N32 TURBINE AND/OR GENERATOR TRIP (UNIT 1)

Reference Type: Off-Normal Procedures
ARI-H13-P680-5 REACTOR CONTROL (LEFT) (UNIT 1)

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-03E

Test T.4.5.3.36, EVALUATION MALFUNCTION SCENARIO #10

Malfunctions Tested: MC01 CONDENSER AIR INLEAKAGE
RD12 SCRAM OUTLET VALVE LEAK
RD15 ATWS
RD18 SDV DRAIN BLOCKAGE
RV03 Relief Valve Fails open
RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

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Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 96% power, EOC conditions, with minor BOP equipment out of service for repair, as well as the HPCS Pump motor. One control rod HCU has low accumulator pressure. The sequence of events for this test are as follows: A stroke of the Main Steam Line drain valves is performed. The operating CRDH Pump trips and the alternate pump is not available. Other withdrawn control rods experience accumulator faults, and reactor power is decreased to perform a manual scram from 50% power. The reactor protection system and alternate rod insertion fail to insert control rods. Actions are taken in accordance with PEI B13 to lower reactor power. Both trains of Standby Liquid Control are started, however the SLC B suction valve fails to open preventing SLC Pump B from starting. Water level is maintained with normal feedwater systems and the main turbine and bypass valves are available for pressure control. Efforts to restore SLC B suction valve are successful by stroking the valve at the MCC. Final plant conditions are reactor power in the source range and decreasing, all rods out, both SLC trains tripped on low storage tank level, with RPV level and pressure under control.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
IOI-8 SHUTDOWN BY MANUAL REACTOR SCRAM
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-N11 MAIN & REHEAT STEAM SYSTEM (UNIT 1)
SOI-N27 FEEDWATER SYSTEM (UNIT 1)
SOI-N33 STEAM SEAL SYSTEM (UNIT 1)
SOI-N34 MAIN LUBE OIL SYSTEM (UNIT 1)
SOI-R10 PLANT ELECTRICAL SYSTEM

Reference Type: Surveillance Procedures
SVI-B21-T2006 MAIN STEAM DRAIN ISOLATION VALVE EXERCISE & STROKE TIME TEST

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-04E

Test T.4.5.3.37, EVALUATION MALFUNCTION SCENARIO #11

Malfunctions Tested: MV04 MOV Failure of auto open circuit
PT01 Process Transmitter Variable failure
RD05 CONTROL ROD ACCUMULATOR FAULT
RD15 ATWS
RD17 LOSS OF CRD PUMP LUBE OIL
RP03 FAILURE OF ARI TO INITIATE
RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 96% power, EOC conditions, with minor BOP equipment out of service for repair, as well as the HPCS Pump motor. The "A" pressure regulator is out of service. The sequence of events for this test are as follows: A surveillance is performed on the Turbine Control System. A failure of the temperature control valve associated with the lead subloop for A RR flow control valve occurs, causing an automatic subloop transfer. The standby subloop starts but fails to restore temperature resulting in a FCV lockup. When a spurious Reactor feedwater pump turbine trip occurs, An automatic Flow control valve runback occurs on B RR loop, A RR loop valve position does not change. The "B" pressure regulator fails downscale, causing all TCVs and BPVs to close. This results in a reactor scram. RPV level is brought under control by operation of the Motor Feed Pump; RPV pressure is controlled by operation of safety relief valves and/or bypass valves using the bypass valve jack. When conditions have stabilized, a non-isolable main steam line rupture occurs in the drywell, causing a rapid decrease in RPV pressure and a rapid degradation of containment, drywell, and suppression pool parameters. Plant Emergency Instruction actions are taken in accordance with PEI's D23-1,2,3, G42, E12, and B13. The Low Pressure Core Spray Pump trips after receiving an automatic start signal. Final plant conditions are as follows; adequate core cooling is assured by either submergence cooling or maintaining reactor pressure 120# greater than containment pressure. Conditions in the containment and drywell are improving due to actions addressed by the PEIs.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
- PEI-B13 REACTOR PRESSURE VESSEL CONTROL
- PEI-D23-1 CONTAINMENT TEMPERATURE CONTROL
- PEI-D23-2 DRYWELL & CONTAINMENT PRESSURE CONTROL
- PEI-D23-3 DRYWELL TEMPERATURE CONTROL
- PEI-E12 SUPPRESSION POOL TEMPERATURE CONTROL
- PEI-G42 SUPPRESSION POOL LEVEL CONTROL

Test T.4.5.3.38, EVALUATION MALFUNCTION SCENARIO #12

Reference Type: Normal Operating Procedures
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-C11(CRDH) CONTROL ROD DRIVE HYDRAULIC SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-P43 NUCLEAR CLOSED COOLING SYSTEM

Reference Type: Periodic Test Procedures
PTI-N32-P0001 TURB OVERSPEED PROT DEVICES TRIP & EHC/TURB LUBE OIL PUMP STARTS/STATOR WATER PUMP START & ROTATIONS WEEKLY TEST

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-05E

Malfunctions Tested: CB01 Spurious breaker trip
PT03 Process Transmitter Variable output clamp
TH14 RECIRC FCV HYDRAULIC POWER UNIT OIL HI TEMP
TH26 MAIN STEAM LINE RUPTURE INSIDE DRYWELL

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to 40% reactor power, with a plant startup in progress. Two APRM channels are inoperable and a half scram signal has been inserted. The sequence of events for this test are as follows: A CRDH Pump shift is performed. After the shift, a failure of two HPCS bistables causes an inadvertent HPCS pump start. The division 3 diesel generator fails to start. The bistables are replaced and the HPCS and Division 3 D/G are returned to standby readiness. When the plant is in a stable condition, Turbine Building Closed Cooling pump A seizes, necessitating starting the standby pump. The common signal to the two RFPT controllers from the master level controller fails, causing the speed of both turbines to increase to their high speed stop. Reactor level rapidly increases to the high level trip setpoint, but the reactor fails to scram. Alternate Rod Insertion (ARI) is successful in partial rod insertion. The Reactor Recirc Pumps are tripped, further lowering power to about 14%. Reactor level is stabilized at 185" by the use of the Motor Feed Pump on the Startup Level Controller. After bypassing the Low Power setpoint (LPSP), manual rod insertion is successful in decreasing power to less than 12%. At that point, the scram and ARI can be reset and the next several manual ARI signals are successful in inserting all control rods. Final plant conditions are the reactor shutdown with RPV level and pressure under control and PEI B13 ready to be exited to ONI C71-1.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Plant Data - Analyses
USAR TABLE 15.5-1 PNPP UPDATED SAFETY ANALYSIS REPORT

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-06E

Test T.4.5.3.39, EVALUATION MALFUNCTION SCENARIO #13

Malfunctions Tested:
BS02 Bistable Spurious trip
CB01 Spurious breaker trip
CN01 Controller Auto/manual failure
CP02 Pump Shaft seizes
DG03 DIESEL GEN SPEED GOVERNOR FAILS
NM04 APRM OUTPUT FAILURE
RD15 ATWS
RD18 SDV DRAIN BLOCKAGE
RY02 Relay Fails as is

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to full power, EOC conditions. Minor BOP equipment is out of service for repair as well as the HPCS Pump Motor. The "B" narrow range level detector is out of service and equalized. The sequence of events for this test are as follows: A bypass valve surveillance is performed. During performance, an SRV cycles open and closed several times until its control power fuses are removed. RHR loop A is placed in the suppression pool cooling mode of operation in order to decrease suppression pool temperature. An RHR pipe break occurs just downstream of the SP suction valve, and the valve cannot be closed. As suppression pool level continues to decrease out the pipe break and into the RHR room and Aux Building (room watertight door jammed open), efforts are attempted to raise SP level with the normal fill mode, the RCIC system on minimum flow, and the Suppression pool makeup system. At a level of 12.2 feet in the pool, the reactor is manually scrammed and an Emergency Depressurization is performed in accordance with Plant Emergency Instructions PEI B13 and PEI G42. Final plant conditions are as follows: the reactor is shutdown with RPV level maintained by high pressure and/or low pressure systems, 8 SRVs are open with the reactor depressurized, and suppression pool temperature is about 145 degrees with level decreasing.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL
PEI-G42 SUPPRESSION POOL LEVEL CONTROL

Reference Type: Normal Operating Procedures
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E51 REACTOR CORE ISOLATION COOLING SYSTEM (UNIT 1)
SOI-G43 SUPPRESSION POOL MAKEUP SYSTEM (UNIT 1)
SOI-P43 NUCLEAR CLOSED COOLING SYSTEM

Reference Type: Off-Normal Operating Procedures
ONI-B21-1 SRV INADVERTENT OPENING/STUCK OPEN (UNIT 1)

Test T.4.5.3.40, EVALUATION MALFUNCTION SCENARIO #14

Reference Type: Surveillance Procedures
SVI-C85-T1314 TURBINE BYPASS VALVE OPERABILITY TEST

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-07E

Malfunctions Tested:
AD01 CYCLING SRV
MV06 MOV Fail as is (mechanical binding)
PC04 SUPPRESSION POOL LEAK
PT01 Process Transmitter Variable failure
RH02 RHR SYSTEM PIPE BREAK

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Evaluation Scenarios used during Initial License training and License Operator Requalification examinations. The scenario is designed to be run in 50 minutes and operator action is required to properly evaluate simulator response. For initial conditions, the simulator is initialized to End of Core Coastdown conditions at the 80% rod line. Minor BOP equipment is out of service for repair as well as the HPCS Pump motor. The suppression pool is at an elevated temperature due to leaking SRVs. ESW and ECC loops "B" are in operation to support the anticipated startup of RHR loop "B" in the suppression pool cooling mode of operation. The sequence of events for this test are as follows: RHR loop "B" is started up in the suppression pool cooling mode. The local division 2 remote shutdown with is bumped, causing a trip of the RHR pump and drainage of the high elevation portions of the piping. After a fill and vent is performed, RHR loop B is returned to standby readiness. A failure of the level transmitter for MSR 1A drain tank occurs. Reactor power is rapidly reduced in accordance with IOI-14 in an attempt to reduce power to less than the scram setpoint for a TCV/TSV closure. The attempt is unsuccessful and a scram signal is generated when the Main Turbine trips due to high MSR level. The reactor fails to scram and PEI B13 is entered in order to shutdown the reactor and stabilize the plant. ARI is unsuccessful in inserting control rods. RPV level is maintained by operation of normal feedwater and pressure is maintained by operation of the bypass valves and SRVs. The Standby Liquid Control system is initiated. Manual control rod insertion with CRDH is successful after bypassing the low power setpoint. Final plant conditions are as follows: Reactor power is less than 4% and decreasing due to operation of SLC and manual rod insertion. All SRVs are closed and suppression pool temperature is decreasing. Normal reactor water level is being maintained.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
IOI-14 FAST UNLOAD & TRIP OF MAIN TURBINE
SOI-B33 REACTOR RECIRCULATION SYSTEM
SOI-C41 STANDBY LIQUID CONTROL SYSTEM (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-P42 EMERGENCY CLOSED COOLING SYSTEM (UNIT 1)
SOI-P45 EMERGENCY SERVICE WATER SYSTEM (UNIT 1)

Test T.4.5.3.41, EVALUATION MALFUNCTION SCENARIO #15

Reference Type: Other
Evaluation Malfunction Scenario OT-3058-ES-08E

Malfunctions Tested: CB01 Spurious breaker trip
PT01 Process Transmitter Variable failure
RD15 ATWS

Discrepancies: Unknown

Evaluators: Later

Test T.4.5.3.42, PEI MALFUNCTION SCENARIO #5 (PART 2)
Revision Number: 00
ANSI/ANS 3.5 Section: 3.1.2 Plant Malfunctions; Other Malfunctions Required to Support Operator Training

Date Tested: Not RUN

Run Time: Later hours

Test Description: The purpose of this test is to verify proper simulator response while conducting Licensed Operator Training in accordance with PEI Malfunction Scenario #5 (part 2). Initial Plant Conditions: the simulator is initialized at 100% power near end of core life. Sequence of events: A complete loss of RPV level instrumentation and failure to Scram causes an entry into PEI-B13. With no indication of level, the reactor is depressurized and flooded. SLC (boron injection) is initiated, but a failure in that system results in the reactor remaining at power. Conditions at end of test: The reactor remains at power, actions are being taken to keep reactor power as low as practical. RPV level is unknown but adequate core cooling is being maintained. Plant recovery is not performed during this scenario.

Baseline Data used For Reference

Reference Type: Emergency Operating Procedures
PEI-B13 REACTOR PRESSURE VESSEL CONTROL

Reference Type: Normal Operating Procedures
SOI-C51(IRM) INTERMEDIATE RANGE MONITORING SYSTEM (UNIT 1)
SOI-C71 RPS POWER SUPPLY DISTRIBUTION (UNIT 1)
SOI-E12 RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)
SOI-E21 LOW PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-E22A HIGH PRESSURE CORE SPRAY SYSTEM (UNIT 1)
SOI-M51/56 COMBUSTIBLE GAS CONTROL SYSTEM & HYDROGEN IGNITERS (UNIT 1)

Reference Type: Other
PEI Malfunction Scenario OT-3034-05A

Malfunctions Tested:
AV02 Air Operated Valve Fails closed
BS01 Bistable Fails to trip
BS02 Bistable Spurious trip
PT01 Process Transmitter Variable failure
RD15 ATWS
RY02 Relay Fails as is
SL05 SLC INJECTION PIPING LEAK

Discrepancies: Unknown

Evaluators: Later

Four Year Test Schedule

Below is a list of the performance tests which will be conducted annually. Pages 2 through 5 list (by year) the tests that will be conducted in years 1 through 4 after filing Form 474. These schedules represent performance of approximately 25% of total tests required for certification per year. Any deletions or changes to the following schedule will require refiling of Form 474 as will changes to the scope or ANSI/ANS 3.5 acceptance criteria related to these tests.

1.0 ANNUAL TESTING

A. Computer Real Time Test

1. T.2.7.1 - Spare Time Verification

B. Steady State Performance Tests

1. T.4.4.1.1 - 25% Power Heat Balance
2. T.4.4.1.2 - 50% Power Heat Balance
3. T.4.4.1.3 - 75% Power Heat Balance
4. T.4.4.1.4 - 100% Power Heat Balance
5. T.4.4.1.5 - 100% Power Stability Test

C. Benchmark Transient Tests

1. T.4.4.2.01 - Manual Scram
2. T.4.4.2.02 - Simultaneous Trip of All Feedwater Pumps
3. T.4.4.2.03 - Simultaneous Closure of All MSIV's
4. T.4.4.2.04 - Simultaneous Trip of All Recirc Pumps
5. T.4.4.2.05 - Single Recirc Pump Trip
6. T.4.4.2.06 - Main Turbine Trip without Reactor Scram
7. T.4.4.2.07 - Maximum Rate Power Ramp (100% - 75% - 100%)
8. T.4.4.2.08 - Maximum Size LOCA with Loss of Offsite Power
9. T.4.4.2.09 - Maximum Size Unisolable Main Steam Line Rupture
10. T.4.4.2.10 - Simultaneous Closure of MSIV's with Single Stuck Open Safety/Relief Valve

Four Year Test Schedule (continued)

2.0 PERIODIC TESTING

A. Year 1 (June 29, 1991 through June 28, 1992)

1. Core Performance Tests

a. T.4.2.1 - Reactor Core Xenon Transient Test

2. Normal Plant Evolutions

a. T.4.3.1 - Cold Shutdown To Reactor Critical
b. T.4.3.2 - Reactor Critical To Turbine Synchronized

3. Transient Tests

a. T.4.5.1.02.AN01 - Annunciator Input Optical Isolator Failure
b. T.4.5.1.06.ED17 - Loss of 125V DC Distribution Panel
c. T.4.5.1.12.IA01 - Air Receiver Leak
d. T.4.5.1.17.NM01 - SRM Detector (pre-amp) Failure
e. T.4.5.1.18.OG03 - Off Gas System Leak Upstream Adsorbers
f. T.4.5.1.22.RH02 - Residual Heat Removal System Pipe Break
g. T.4.5.1.27.TH01 - Recirc Loop Rupture (DBA LOCA)
h. T.4.5.1.27.TH27 - Main Steam Line Rupture In Steam Tunnel

4. Malfunction Scenarios

a. T.4.5.3.01 - Loss of Feedwater Heating (RRFC In Auto)
b. T.4.5.3.02 - Loss of Feedwater Heating (RRFC In Manual)
c. T.4.5.3.03 - Feedwater Controller Failure-Maximum Demand
d. T.4.5.3.04 - Pressure Regulator Failure-Open
e. T.4.5.3.22 - PEI Malfunction Scenario #1
f. T.4.5.3.23 - PEI Malfunction Scenario #2
g. T.4.5.3.24 - PEI Malfunction Scenario #3
h. T.4.5.3.25 - PEI Malfunction Scenario #4

Four Year Test Schedule (continued)

B. Year 2 (June 29, 1992 through June 28, 1993)

1. Core Performance Tests

   a. T.4.2.2 - Core Flux Distribution Test
   b. T.4.2.5 - Core Thermal Performance Test

2. Normal Plant Evolutions

   a. T.4.3.3 - Power Increase To 100% Power

3. Transient Tests

   a. T.4.5.1.04.CU03 - Reactor Water Cleanup System Pipe Break Outside Containment (Steam Tunnel)
   b. T.4.5.1.07.EG01 - Main Generator Lockout Relay Trip
   c. T.4.5.1.12.IA02 - Instrument Air Line Leak
   d. T.4.5.1.17.NM02 - IRM Detector (pre-amp) Failure
   e. T.4.5.1.18.OG04 - Off Gas System Leak Downstream Adsorbers
   f. T.4.5.1.23.RP02 - Inadvertent Initiation of Alternate Rod Insertion
   g. T.4.5.1.27.TH19 - RPV Level Inst Reference Leg Break
   h. T.4.5.1.28.TU01 - Main Shaft Oil Pump Degradation

4. Malfunction Scenarios

   a. T.4.5.3.05 - Inadvertent Safety Relief Valve Opening
   b. T.4.5.3.06 - Inadvertent RHR Shutdown Cooling Operation
   c. T.4.5.3.07 - Pressure Regulator Failure-closed
   d. T.4.5.3.08 - Generator Load Reject With Bypass Valves
   e. T.4.5.3.09 - Generator Load Reject Without Bypass Valves
   f. T.4.5.3.10 - Turbine Trip
   g. T.4.5.3.11 - Loss of AC Power (Loss of Aux Transformer)
   h. T.4.5.3.26 - PEI Malfunction Scenario #5 (part 1)
   i. T.4.5.3.27 - Evaluation Malfunction Scenario #1
   j. T.4.5.3.28 - Evaluation Malfunction Scenario #2
   k. T.4.5.3.29 - Evaluation Malfunction Scenario #3
   l. T.4.5.3.30 - Evaluation Malfunction Scenario #4
   m. T.4.5.3.31 - Evaluation Malfunction Scenario #5

Four Year Test Schedule (continued)

C. Year 3 (June 29, 1993 through June 28, 1994)

1. Core Performance Tests

   a. T.4.2.3 - Core Thermal Power vs. Recirc Flow Test
   b. T.4.2.4 - Core Flux Response to Rod Movement

2. Normal Plant Evolutions

   a. T.4.3.4 - Power Decrease to Turbine/Generator Unloaded

3. Transient Tests

   a. T.4.5.1.06.ED06 - Loss of 480V Bus
   b. T.4.5.1.09.FW02 - Feedwater System Pipe Break Inside Drywell
   c. T.4.5.1.15.MC01 - Condenser Air Inleakage
   d. T.4.5.1.17.NM03 - LPRM Detector Failure
   e. T.4.5.1.21.RD03 - Control Rod Drift - In
   f. T.4.5.1.25.SW01 - Nuclear Closed Cooling System Process Piping Leakage
   g. T.4.5.1.27.TH20 - RPV Level Inst Variable Leg Break

4. Malfunction Scenarios

   a. T.4.5.3.12 - Failure of RHR Shutdown Cooling
   b. T.4.5.3.13 - Recirc Flow Control Failure-decreasing (both Flow Control Valves)
   c. T.4.5.3.14 - Recirculation Pump Seizure
   d. T.4.5.3.15 - Abnormal Startup of Idle Recirculation Pump
   e. T.4.5.3.16 - Recirculation Pump Shaft Shear
   f. T.4.5.3.32 - Evaluation Malfunction Scenario #6
   g. T.4.5.3.33 - Evaluation Malfunction Scenario #7
   h. T.4.5.3.34 - Evaluation Malfunction Scenario #8
   i. T.4.5.3.35 - Evaluation Malfunction Scenario #9
   j. T.4.5.3.36 - Evaluation Malfunction Scenario #10

Four Year Test Schedule (continued)

D. Year 4 (June 29, 1994 through June 28, 1995)

1. Core Performance Tests

a. T.4.2.6 - Core Subcritical Multiplication Test
b. T.4.2.7 - Reactor Core Life Test
c. T.4.2.8 - Shutdown Margin Demonstration

2. Normal Plant Evolutions

a. T.4.3.5 - Plant Cooldown to Cold Shutdown

3. Transient Tests

a. T.4.5.1.06.ED09 - Loss of 125V DC Bus
b. T.4.5.1.09.FW03 - Feedwater System Pipe Break Outside Containment
c. T.4.5.1.16.MS11 - Steam Seal Header Pressure Regulator Failure
d. T.4.5.1.17.NM10 - Neutron Monitoring Detector Drive Stuck
e. T.4.5.1.21.RD04 - Control Rod Drift - Out
f. T.4.5.1.25.SW02 - Service Water System Process Piping Leakage
g. T.4.5.1.27.TH21 - Power/Flow Instabilities (IEB 88-07 Supplement 1)

4. Malfunction Scenarios

a. T.4.5.3.17 - Recirc Flow Control Failure-increasing (both Flow Control Valves)
b. T.4.5.3.18 - Control Rod Drop Accident
c. T.4.5.3.19 - Inadvertent High Pressure Core Spray Startup
d. T.4.5.3.20 - Anticipated Transient Without Scram (ATWS)
e. T.4.5.3.21 - Loss of Off-Site Power
f. T.4.5.3.37 - Evaluation Malfunction Scenario #11
g. T.4.5.3.38 - Evaluation Malfunction Scenario #12
h. T.4.5.3.39 - Evaluation Malfunction Scenario #13
i. T.4.5.3.40 - Evaluation Malfunction Scenario #14
j. T.4.5.3.41 - Evaluation Malfunction Scenario #15
k. T.4.5.3.42 - PEI Malfunction Scenario #5 (part 2)