SHEARON HARRIS NUCLEAR POWER PLANT

OPERATOR TRAINING SIMULATOR

INITIAL SIMULATOR CERTIFICATION

MARCH 15, 1991

CAROLINA POWER & LIGHT COMPANY
NEW HILL, NORTH CAROLINA

SHNPP CERTIFICATION PACKAGE

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

U.S. NUCLEAR REGULATORY COMMISSION

SIMULATION FACILITY CERTIFICATION

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.

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: Shearon Harris Nuclear Power Plant - Unit 1

DOCKET NUMBER: 50- 400

LICENSEE: Carolina Power and Light Company

DATE: 03/15/91

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 [X] and describe fully on additional pages as necessary.

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

Harris Simulator - Harris Energy & Environmental Center
Route 1, Box 327
New Hill, North Carolina 27562

[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)*

See Section 4.0, "Simulator Tests", and Appendix F, "Simulator Certification Test Abstracts."

[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)*

See Section 4.0, "Simulator Tests;" Appendix A, "Schedule of Annual Operability Tests;" and Appendix B, "Schedule of Malfunction Tests."

[ ] 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)*

NOT APPLICABLE - INITIAL CERTIFICATION

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

NOT APPLICABLE - INITIAL CERTIFICATION

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: [signature]

TITLE: Vice President, Nuclear Services Department

DATE: 3-21-91

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

INTRODUCTION

General Information

The Shearon Harris Nuclear Power Plant Simulator Certification Package is provided to demonstrate compliance with the requirements of 10CFR55.45(b) including compliance with ANSI/ANS-3.5-1985 as implemented by NRC Regulatory Guide 1.149-1987. The subject simulator facility consists solely of a plant referenced full scope simulator, which is the primary vehicle for providing positive, practical license training. A formal training value assessment prepared by the Harris Training Unit was presented to Harris Plant Operations for their concurrence and was signed by the Operations Manager. The documentation provided herein is intended to constitute sufficient basis for the certification of the Harris Simulator.

Simulator Configuration Control Board (Simulator Review Group)

The primary means of evaluation and review of the simulator operations is the Simulator Review Group (SRG). The SRG performs the same function as the Simulator Configuration Control Board. This group is made up of Plant Operations, Operations Training, and Simulator Support Personnel. The SRG will include at least one degreed engineer, one currently licensed Harris Plant operator, and one SRO licensed or certified simulator instructor. The group reviews proposed non-routine changes to the simulator, such as changes to the scope of simulation or any desired changes in simulator capability. These evaluations will be documented as training value assessments. The SRG reviews outstanding simulator certification discrepancies for their impact on training so that high priority items can be identified. The SRG reviews differences between the simulator and the plant to ensure they do not detract from training. Minutes of SRG meetings are maintained to serve as a record of SRG decisions. The current SRGs' qualifications are included as Appendix D.

EXCEPTIONS TO ANSI/ANS-3.5

The exceptions identified during certification testing or the review/analysis of ANSI/ANS-3.5 are contained in this section of the submittal package. The exceptions are listed by ANSI-3.5 reference and subject. Each specific exception taken and its associated justification is addressed on an item by item basis and was reviewed and approved by the Simulator Review Group to ensure the exceptions do not detrimentally impact the license operator training program and do not prevent 10CFR55 compliant simulator examinations (operating tests) from being conducted.

1. ANS Section 3.1.1(7)--Operations at less than full RCS flow

This section is not applicable. Power operations with less than three operating reactor coolant pumps is prohibited by Technical Specifications. However, the simulator is capable of such operations.

2. ANS Section 3.1.1(9)--Core Performance Testing

Rod worth and reactivity coefficient measurement procedures were not performed for certification testing. These tests are performed by Technical Support Unit, not operators. Tests which were conducted applicable to this section were Estimated Critical Conditions, Shutdown Margin, and Heat Balance.

3. ANS Section 3.1.2(12)--Control Rod failures

Drifting rods are not simulated as this type of failure is not relevant to the rod mechanisms used at SHNPP.

Uncoupled rods are not simulated in the SHNPP simulator. This malfunction was evaluated by the Simulator Review Group and a Training Value Assessment deemed uncoupled rods as not needed for effective operator training due to its low probability of occurrence.

4. ANS Section 3.1.2(25)--Reactor Pressure Control System Failure including Turbine Bypass Failure (BWR)

This item is specifically related to Boiling Water Reactors.

5. ANS Section 3.2.3--Control Room Environment (Communication's Systems)

The sound-powered phone circuits used in the control room are not currently simulated. The Simulator Review Group is in the process of conducting a cost-benefit analysis and assessing the training value of adding this system. The evaluation will be complete by April 30, 1991. A radio system simulation is currently used in lieu of the sound-powered phones.

6. ANS Section 3.2.3--Control Room Environment (Ceiling and Lighting)

The current ceiling and lighting are approximately 20 feet above the simulator panels rather than 3 feet as in the plant to facilitate visitor viewing of the simulator from above. Lowering of the ceiling and modifying the lighting to fail based on the simulation of electrical bus failures is currently under management review. A decision on these changes will be made by June 30, 1991.

7. ANS Section 4.1(3)--Steady State Accuracy Tests (critical parameters)
ANS Section 4.1(4)--Steady State Accuracy Tests (non-critical parameters)

The criteria used for the comparison between the simulator and plant parameters was 2% (10% for non-critical parameters) of the associated instrument loop range. In addition, the parameter variation must not detract from training. The standard states to use 2% (10%) of the reference plant parameter. Using a percentage of the instrument loop range more realistically represents the difference which can be noted by the operators and is the method used in the draft ANSI/ANS-3.5 of 1990. This method was reviewed and approved by the Simulator Review Group.

8. ANS Section Appendix B.1--BWR Simulator Operability Test

This item is specifically related to Boiling Water Reactors

9. ANS Section B.2.1(2)--Steady State Data

Steam generator temperature was not measured as this parameter is only applicable to once-through type steam generators.

10. ANS Section 3.1.2(1)(c)--Large and Small Reactor Coolant Breaks

ANS Section B2.2(8)--Maximum Size Reactor Coolant System Rupture Combined with Loss of All Off-site Power

ANS Section B2.2(10)--Slow Primary System Depressurization to Saturated Condition Using Pressurizer Relief or Safety Valve Stuck Open

Due to current problems with the RCS model, certain RCS break malfunctions do not respond properly. Exact discrepancies are noted in the test abstracts. The RCS model is scheduled for upgrade to correct these deficiencies by December 31, 1992.

1.0 Simulator Information

1.1 Simulator General

1.1.1 Owner: Carolina Power & Light Company
1.1.2 Referenced Unit: Shearon Harris Nuclear Power Plant, Unit #1, Westinghouse 3-Loop PWR
1.1.3 Simulator Supplier: Westinghouse Electric Corporation
1.1.4 Ready-for-Training Date: December 20, 1985
1.1.5 Type of Report: Initial Certification

1.2 Simulator Control Room

1.2.1 Physical Arrangement: The simulator control room is approximately 80% as large as the referenced unit. The simulator control room panels are the same size as the referenced unit, but some of the auxiliary panels have been moved or angled slightly to accommodate space restrictions and the protrusion of the instructor station area into the simulator area. All simulated panels are in the same relative location as the referenced plant control room and provide the same visual perspective as in the plant. Figures 1 and 2 are provided for simulator to plant layout comparisons at the end of this section.

Some other minor differences exist such as the carpet color, the location of handrails, the type of furniture installed, and the size and exact shape of some information/status boards. Differences between the simulator and the control room which will remain have been evaluated by the Simulator Review Group as acceptable.

1.2.2 Panels and Equipment: All control room panels are included in the simulation except the Condensate Booster Pump Panel, Seismic Monitoring Panel, and the Digital Metal Impact Monitoring Panel. These three panels are omitted based on training value assessment.

With the exception of Emergency Response Facility Information System (ERFIS) peripherals no panels outside the control room are included in the simulation facility. A second training value assessment is being performed on the Digital Metal Impact Monitoring Panel based on changes to operating procedures. Classroom and on-the-job training are the means to provide training on this system.

The communications equipment provided in the simulator are nearly identical to those in the control room. The phone system is set up such that the most commonly called numbers can be dialed from the simulator floor using the same numbers as in the plant. All dialed numbers will ring in the simulator instructor's booth. A light on the phone will show the instructor which number has been dialed. In addition, the radio console is modelled on the simulator and it calls the instructor booth also. The radio beeper system, which was recently added at the plant, and the sound-powered phone system (see Exception #5) are not currently included in the simulation facility. A cost-benefit analysis is being performed to evaluate the desirability of including those systems in the simulator.

1.2.3 Systems: All operative plant systems assessable from the control room are simulated except for Seismic Monitoring, Digital Metal Impact Monitoring, and Waste Processing. These systems are omitted based on training value assessment.

1.2.4 Environment: Some differences exist in the ceiling, lighting, and sound environment between the simulator and reference plant control rooms (see Exceptions #5 and #6). These differences are scheduled for resolution before the end of 1992. None of the differences have significant training impact.

The simulator control room is designed to include a viewing platform for visitors to the Harris simulator. This results in a difference between the simulator and main control room overhead and lighting. Recommendations concerning whether the simulator ceiling will be lowered and the lighting will be changed to more closely replicate the control room are under management review.

1.3 Simulator Instructor Interface

1.3.1 General Instructor System: The Harris Simulator has an instructor's booth that is separated from the simulator control room and out of sight (one way mirrored glass) from the operator's view. The instructor is able to observe the actions of the operators in the simulator control room from the booth. An audio (microphone)/video (camera) system is provided in the simulator facility to allow better analysis of operator activity. The audio/video system has been reviewed by the SRG and deemed acceptable as a no-training impact difference.

The instructor controls all the functions of the simulator from the control booth. This is accomplished by using a control board and three control and monitoring CRT's. Using the control boards and CRT's, the instructor establishes simulator initial conditions, inputs operating malfunctions, simulates local operator actions, and is also able to interact with students. The instructor is also able to monitor most plant parameters from the booth.

The instructor has the capability of operating the simulator from the instructor's booth or from the simulator control room using a hand held remote operating device.

1.3.2 Initial Conditions: After the simulator is started up by the instructor, there is the possibility of initializing the simulator from 50 initial conditions (ICs). The first 20 ICs are stabilized and resnapped after each major simulator modification/upgrade period but prior to training restart. These first 20 ICs contain a minimum of 3 power levels at 3 times in core life (BOL, MOL, and EOL), hot standby, and other primary training starting points as selected by the trainers. The description of the 50 ICs is as follows:

* ICs 1 through 20 are permanent ICs that are password protected.
* ICs 21 through 40 are instructor-developed/maintained ICs, which may, or may not, be password protected.
* ICs 41 through 48 are dedicated to simulator maintenance activities.

* IC 49 is a default snapshot IC.
* IC 50 is a default IC to allow the instructor to return to interrupted scenario at the point it was interrupted.

After selecting an IC, the simulator is selected to RUN to commence operation at real time.

Simulator Support Procedure SSP-215.4 details the administrative requirements for control of the IC sets.

1.3.3 Malfunction Selection: Stored within the simulator is a wide array of plant malfunctions ranging from major casualties to more isolated equipment failures. To select a particular malfunction, the instructor can input the malfunction number directly or select it from a menu driven malfunction list for selected systems.

The instructor selects the time at which the malfunction is to start. An unlimited number of malfunctions can be directed to occur at different times or, if desired, all at the same time. In addition, for analog malfunctions, the instructor can vary the severity of the malfunction and a ramp time over which the malfunction is to be implemented. Malfunctions can also be initiated based on a particular plant condition occurring.

The status of any selected malfunction can be determined at any time from the instructor's console. This includes the delay times, the severity factor, and the ramp time.

1.3.4 Simulator Overrides

1.3.4.1 Panel Overrides

The instructor has the ability to override any device on the main control board panels. For example, a meter may be driven to any value, a light may be turned off or on, or a switch may be failed closed. The override may be inserted with a time delay, and analog values may be ramped in over a specified time band.

1.3.4.2 Transmitter Overrides

As part of the continuing enhancement of the simulator, a transmitter override feature is being added to applicable modelled transmitters. The transmitter can be driven to any value in it's range so that, if provided, corresponding bistable trips and automatic actions will occur. The override may be ramped in over a specified time period if desired by the instructor. Thus far, overrides for 101 transmitters have been implemented.

1.3.5 Local Operator Actions (LOAs): To add to the realism of the actual plant operations, the simulator has the capability of simulating LOAs that alter plant operation and control room indications. An example of a LOA is the tripping of a back panel bistable with status light indication in the control room which changes the coincidence for actuation of a reactor trip. LOAs are used to perform those plant manipulations which cannot be performed from the control room, such as operating manual valves, etc.

LOA's are selected by the instructor from a menu or from direct input into the computer.

1.3.6 Parameter and Equipment Monitoring

1.3.6.1 Plant Parameter Status Display: One CRT in the instructor's booth is capable of providing a "log sheet" of the status of plant parameters, such as reactor power, RCS temperature, pressurizer pressure, etc.

1.3.6.2 Equipment Status Display: A CRT in the instructor's booth allows the selection of certain systems, such as CVCS, to determine the status of the various pumps, valves, etc., in that system. The status is indicated as "on/off" or "open/closed."

1.3.6.3 Parameter Versus Time Plots: Using the CRT in the instructor's booth, the instructor is able to monitor up to four variables using vector graphics. The instructor selects the various parameters to be plotted. X-Y plots are also facilitated to plot one parameter against another.

1.3.7 Simulator Special Features

1.3.7.1 Switch Check Status: The switch check feature of the simulator allows the instructor to ensure the proper positioning of control room switches and potentiometers for each IC. A light on, or very near, a switch or potentiometer will blink if it is not in the proper position for the selected IC.

The instructor is able to review the status of all switches and potentiometers on the CRT when the SWITCH CHECK STATUS function is selected.

As an added feature, the instructor can override the switch check if the out-of-position switch is of little significance to the evolution to be run.

1.3.7.2 Simulator Freeze Function: When this function is selected, the simulator is frozen. The conditions of the simulator at that point may then be snapped into an IC if desired. The instructor can restart the simulation by placing the simulator in RUN.

1.3.7.3 Backtrack Function: The backtrack function allows the instructor to back up and restart the simulator from some time since the last IC was restored.

1.3.7.4 Snapshot: The snapshot feature of the simulator enables the instructor to save a condition at any given point of an evolution. The instructor can then use the snapshot as an IC when desired. This includes saving the malfunctions that were selected. The snapshot is stored in a default IC.

1.3.7.5 Replay: The replay feature provides the instructor with the capability of freezing the simulator and to replay as much as 120 minutes prior to that point. This allows students to review trends in parameters and the effect that their actions had on plant operations. This feature is currently disabled to minimize impact on real time performance.

1.3.7.6 Fast Time: The fast time feature allows the instructor to accelerate through plant evolutions to a point desired by the instructor.

The following list of plant evolutions can be selected from the menu when the FAST TIME push button is depressed:

* Fission product concentration and decay heat
* RCS heatup
* RCS cooldown
* Turbine system heatup
* Turbine system cooldown

When the FAST TIME push button is selected, the dynamics of the selected plant parameters will speed up while all other parameters remain at real time.

1.3.7.7 Slow Time: The slow time feature of the simulator allows the instructor to slow the dynamics of a particular evolution or scenario. The students are then able to evaluate trends, parameters, etc., that may not be observable in real time operation.

When the SLOW TIME push button is depressed, all plant dynamics are slowed to a preselected lower frequency.

1.3.7.8 Automatic Exercise: The automatic exercise feature of the simulator allows the selection of up to 20 preprogrammed lesson drills which will automatically step the simulator through a set of predefined operations and controls. This feature minimizes the setup and manipulations required of the instructor by providing standard, repeatable, and planned exercises on the simulator.

1.3.7.9 Simulator Operating Limits - In accordance with ANSI/ANS-3.5 section 4.3, the simulator will alert the instructor by way of a message displayed on the control CRT, if any number of operating limits are exceeded which could lead to negative operator training, or indicate that simulation is proceeding out of the limits of model design. The limits used to alert the instructor are as follows:

- Containment Temperature > 400 degrees
- Containment Pressure > 60 psia
- RCS Pressure > 2700 psia
- Thermocouple Temperature > 1550 degrees
- RCS Boron > 2500 ppm
- RCS Boron < 0 ppm
- Steam Generator Pressure > 1400 psia
- Steam Generator Steam Flow > 6 MLBH
- Core Power > 120 %
- Condenser Pressure > 20 psia

1.4 Operating Procedures for the Reference Plant

The Simulator Control Room utilizes a controlled set of procedures identical to those used in the reference unit's control room.

1.5 Identify Changes Since Last Report

No changes - initial report.

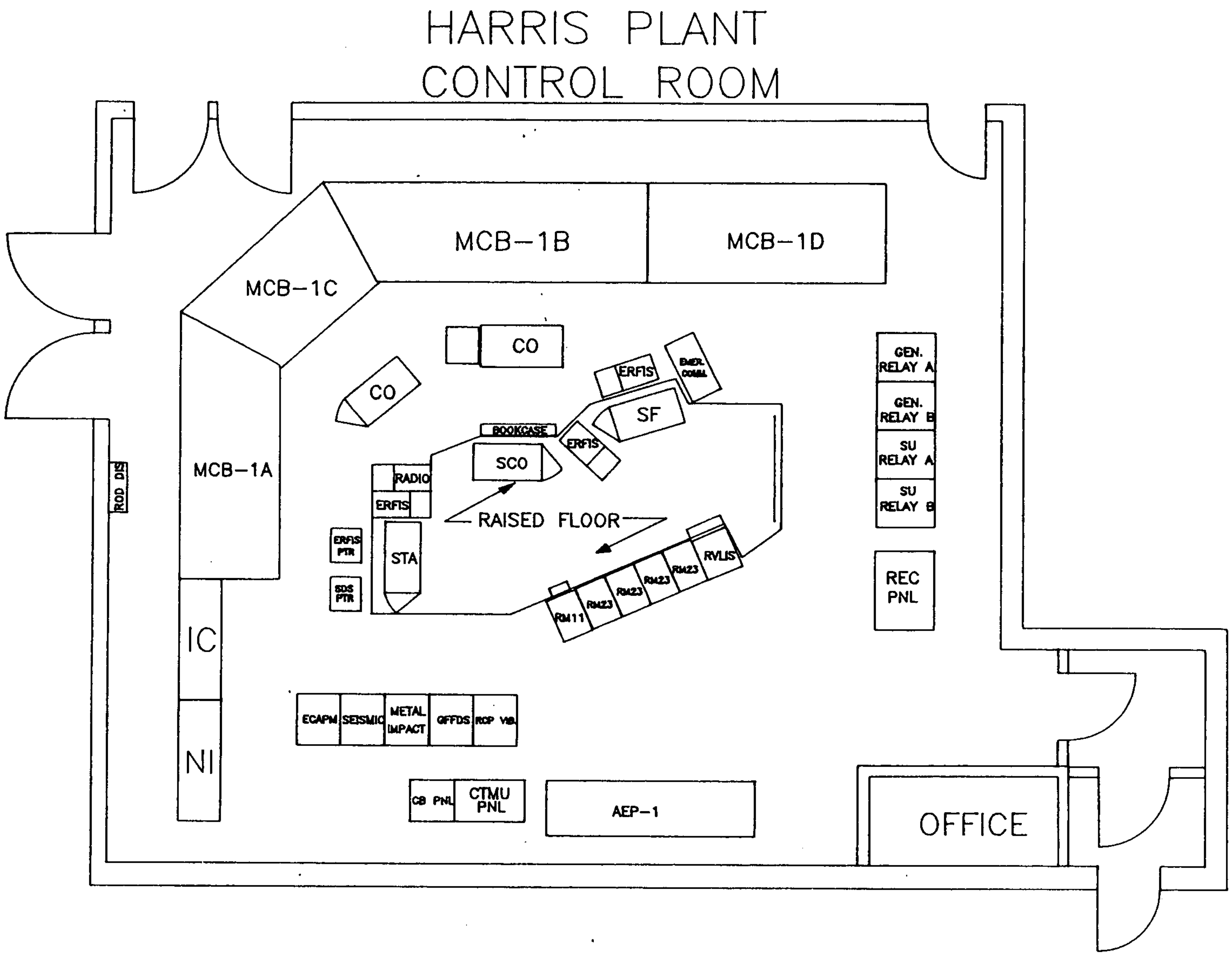

Figure 1: Plant Control Room Layout

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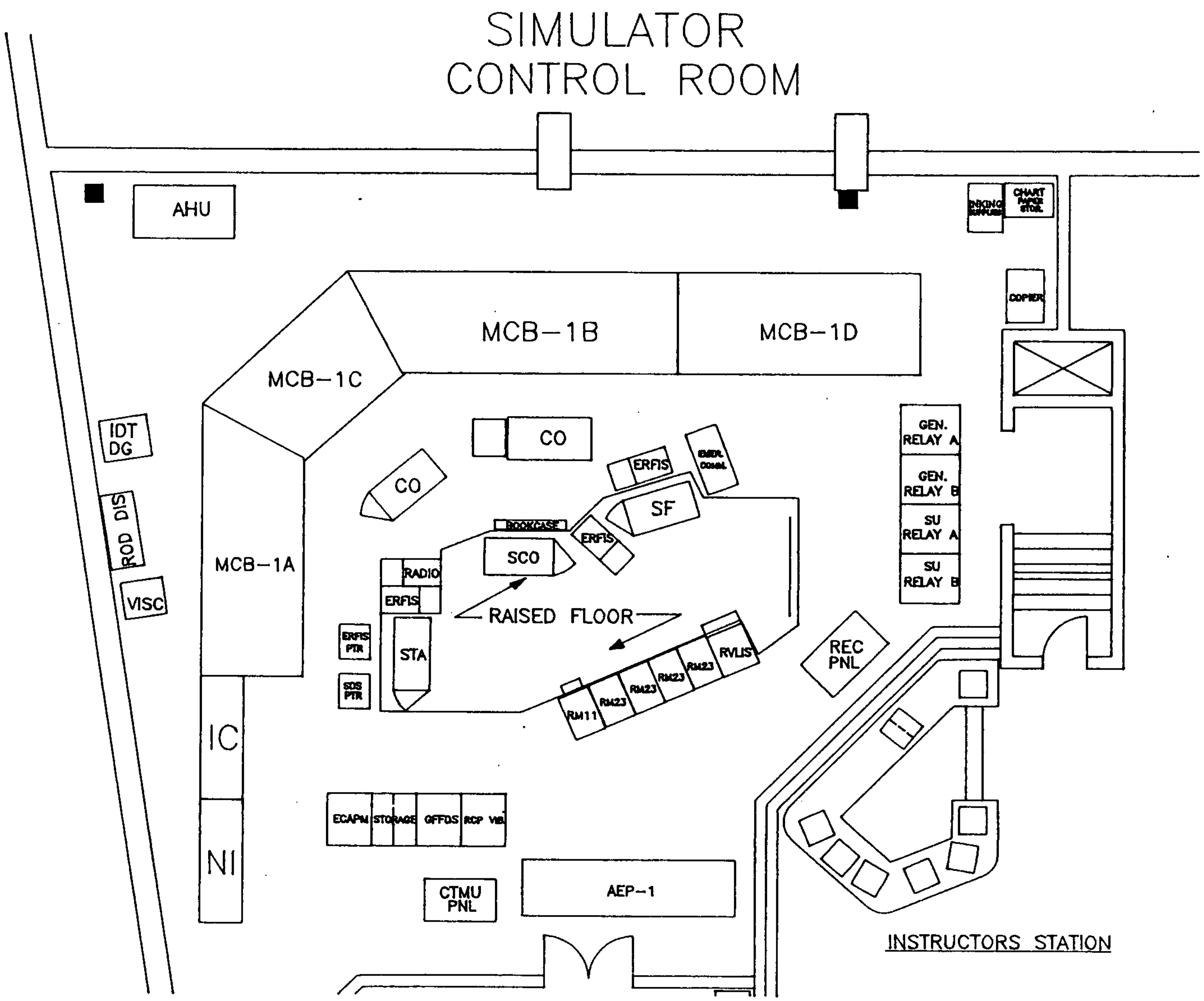

Figure 2: Simulator Control Room Layout

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2.0 Simulator Design Data Base

The original simulator design data base consists of plant reference drawings (logics, CWDs, P&IDs, etc.), FSAR, Plant Operating Manuals (POMs) including system descriptions, and system test results which were sent to the simulator vendor for simulator construction. A complete set of these reference documents is available to the Harris Training Unit for use by the simulator support staff in simulator modification, troubleshooting, and updating. The design data base was submitted to the vendor prior to actual startup of the Harris Plant. Updated design data obtained since the Harris Plant has been operational is used to perform simulator modifications. This design data is maintained as part of the Simulator Update Design Data.

The original data base was updated with additional plant modification data prior to simulator delivery. This data brought the simulator to a plant design point of April 1985 as of delivery. Plant modification/change data have continued to be collected and analyzed for simulator applicability through formally controlled distribution of Plant Change Request (PCR's), documentation updates, and plant procedure changes.

Routine changes, which are defined as those that can be implemented using existing budgeted resources and that do not change the scope of simulation, are approved by virtue of approval of the associated plant change. Non-routine changes, which are defined as those changes that require resources beyond normal support activities or that do result in a simulator scope change, are presented to the Simulator Review Group (SRG) for review and approval. Those changes rejected by the SRG are rejected on the basis of a training value assessment, which would include a cost benefit analysis.

The Manager - Simulator is an active member of the Plant Review Group (PRG). This activity facilitates improved simulator visibility to corporate engineering and plant maintenance and modification organizations which, in turn, results in simulator issues being properly considered along with plant modification projects. The PRG reviews plant modifications early in the development stage, thereby giving the simulator support organization advanced information needed to plan resultant simulator changes.

3.0 Simulator Discrepancy Resolution and Upgrading Programs

3.1 Simulator Trouble Report System

Discrepancies noted in the simulator during testing or training sessions will be documented by a simulator Trouble Report (TR). The TR is one of two categories identified on the Simulator Service Request (SSR) form. The other category is potential change, which is used to initiate consideration of desired simulator enhancements. Persons noting a problem in the simulator may submit a simulator Trouble Report (TR). In addition, the simulator staff will generate a Trouble Report as a response to trainee feedback. The TR will describe the problem noted with sufficient information for the simulator staff to identify the problem for resolution. Trouble Reports which are determined by the Manager - Simulator to be valid and require modification to the simulator, are used to generate Simulator Service Requests as described in section 3.3. If a TR has questionable value or would require a change in the scope of simulation, it is reviewed by the Simulator Review Group for resolution.

3.2 Plant Change Request Implementation

All Plant Change Requests (PCRs) which are approved for work and which have the potential to impact the simulator, are reviewed by the simulator operations staff for applicability to the simulator. This review is a continuous process and PCRs are reviewed by the simulator staff in the same time frame during which they are presented to the plant staff. PCR's which are within the current simulator scope and have an effect on the simulator are assigned a Simulator Modification Request (SMR) in accordance with Simulator Support Procedure (SSP)-217.2, Simulator Modification Requests/Work Orders. PCR's which may have an impact on simulator training, but which are currently outside the simulator scope are reviewed by the Simulator Review Group (SRG) to determine training value. If the SRG decides that a PCR should be implemented in the simulator, an SMR is generated to have the modification included.

SMRs are scheduled to be completed in the simulator within twelve months of their completion in the plant. If requested by the plant operations staff, the modification may be performed in the simulator prior to its completion in the plant in order that the operators may be trained prior to plant modification completion.

Actual modification to the simulator hardware or software is accomplished in accordance with SSP-217.3, Modification. Reference documents and other information used to perform the modification are filed with the completed SMR package and kept as part of the simulator Update Design Data Base.

Testing of the modification prior to its turnover is accomplished in accordance with SSP-217.4, Modification Testing.

3.3 Simulator Service Request Program

Trouble Reports which are written as a result of simulator training or testing are used to generate Simulator Service Requests (SSRs). The SSR is used by the simulator operations and software staffs to evaluate the problem, and to identify the corrective action. Documentation used to research the problem is attached to the SSR for inclusion as part of the simulator Update Design Data Base. Modifications which result from an SSR are tested in accordance with SSP-217.4, Modification Testing.

3.4 Simulator Configuration Management System

Simulator Configuration management is the mechanism by which controls are placed on how the simulator is maintained consistent with the referenced plant/unit. A flowchart of the system is provided as Figure 3 at the end of this section. A Personal Computer based automated Configuration Management System is used for recording and tracking Simulator Service Requests and Simulator Modification Requests. Simulator Service Requests are entered into the system when written. The system automatically records the entry date, and assigns a sequential number to the SSR. The initiator checks whether the SSR is for a known problem or a proposed enhancement to the simulator and this information is entered into the system.

After the SSR is entered into the system, it is sent to the Simulator Specialist for disposition. The Specialist assigns a priority number of 1 through 4 to the SSR. SSP-217.5, Work Load Prioritization, uses the same prioritization criteria as the Nuclear Plant Prioritization Process and is used to schedule simulator work. Next, for Trouble Reports, the SSR is pre-tested to insure its validity and repeatability. If a member of the simulator staff initiated the SSR, the pretest may be skipped. The SSR is then assigned to a Software Specialist for work.

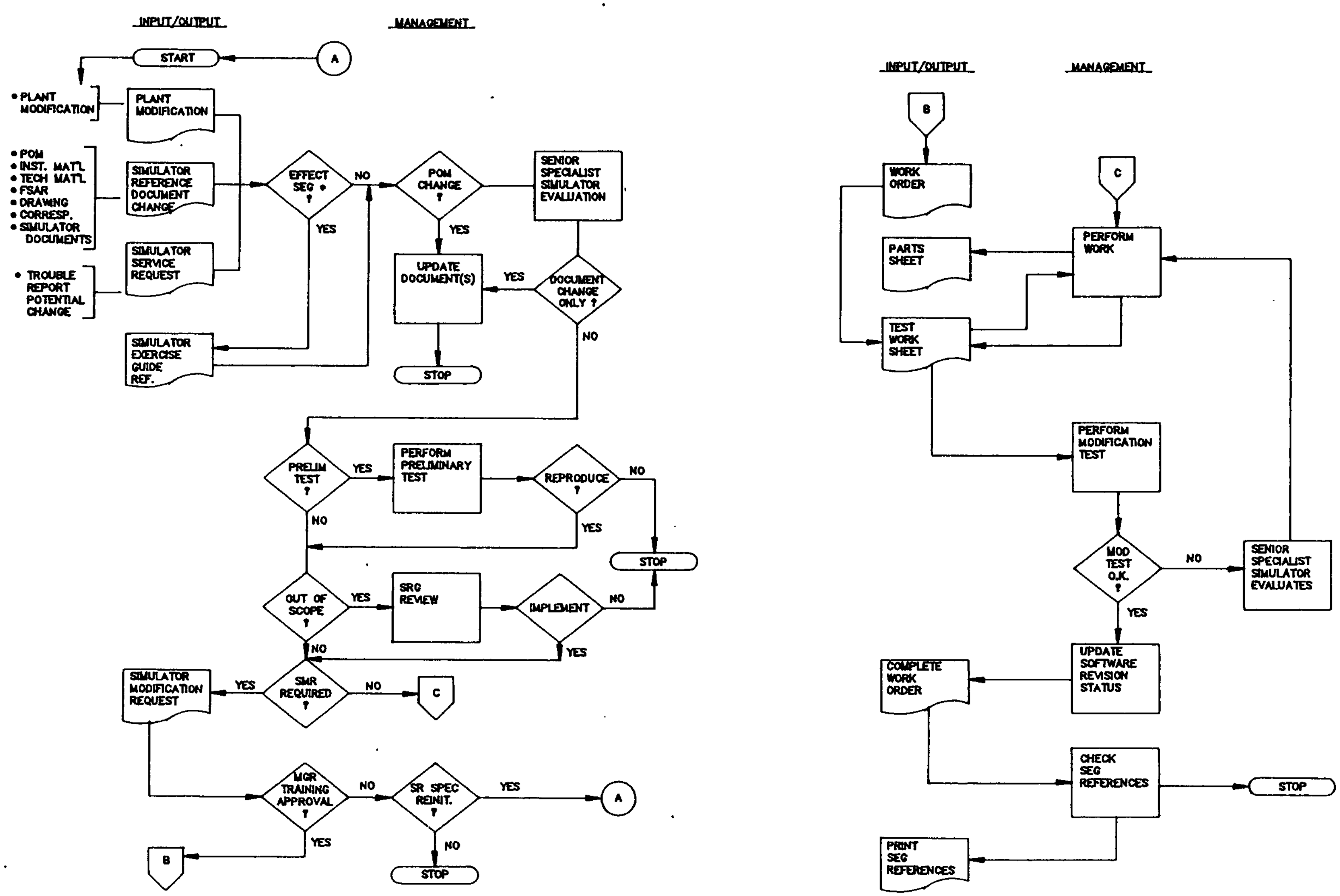
Plant Change Requests are reviewed by the Simulator Support Staff for applicability to the simulator. The PCR is entered into the CMS computer when it is received. When the decision is made as to the PCR's applicability, this information is also entered into the computer. If the PCR is applicable, the CMS computer will automatically generate a Simulator Modification Request. In addition, it will automatically generate Simulator Work Orders which are assigned to the software and hardware personnel as applicable.

During the development of simulator software modifications, the support staff uses a development simulator system. This system runs independently of the training simulator system, and contains all the software modules. This allows software development and preliminary testing to be conducted simultaneously with training activities. The simulator support staff works to clear several SSR's and SMR's at one time. When several modifications are ready and testing time is available on the simulator, the development disk will become the training disk and final tests will be run with the new software load. There are actually three disks in use for the simulator. One is the previous training disk, one is the current training disk, and one is the development disk. This facilitates a "fall-back" capability to a previous training load with minimal training schedule impact (< 5 minutes) should a difficulty be encountered which prevents continuance of training on the current training load.

When PCRs or SSRs are completed, their status is updated in the CMS computer. The CMS computer is used to printout weekly, and as other needs arise, reports as to the status of outstanding plant modifications and service requests.

The CMS computer is also used to generate schedules for and track completion of simulator preventive maintenance activities.

CONFIGURATION MANAGEMENT FLOW



* SEG — SIMULATOR EXERCISE GUIDE

Figure 3: Configuration Management Flowchart

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4.0 Simulator Tests

The simulator certification test process was developed, performed, and reviewed by presently or previously NRC licensed or certified individuals.

The selection of simulator performance test topics was determined based on ANSI/ANS-3.5-1985 requirements and a comprehensive review of the licensed operator training program.

Each simulator performance test was developed after a review of available reference material relevant to the selected topic including extensive use of system descriptions, plant procedures, logic diagrams, block diagrams, control wire diagrams, Final Safety Analysis Report, Licensee Event Reports, Significant Operating Event Reports, reference or similar plant data, and appropriate technical manuals. The validity of the tests and test results was determined by a "panel of experts" drawn from the primary members of the SRG and other engineers. (See Appendix D.)

The acceptance criteria for each individual simulator performance test is contained within the actual structure of the test itself. Each simulator performance test contains sign-off lines for each step of the test. If a simulator response was not as specified in a test step, the person performing the test was required to document the deficiencies within the test and also on the associated test abstract.

Deficiencies found during the performance of these tests are listed by system in Appendix E and were reviewed for their impact on training. After this review, recommendations were presented to the Simulator Review Group for their concurrence and approval.

4.1 Certification Test Schedule

4.1.1 Annual Operability Tests - The annual operability tests include the simulator Real Time Test, the Physical Fidelity Test, the Steady State Stability and Accuracy Tests, Normal Operating Tests, Normal Operator Surveillance Tests, and the Transient tests. These tests are listed in Appendix A.

The Normal Operator Surveillance Tests developed for certification are not inclusive of all operator-conducted surveillance testing. The full list of operated-conducted surveillance includes over two hundred tests. For certification, tests were selected which would provide meaningful data for simulator to plant comparison, and which had a high probability of being incorporated into operator training programs.

4.1.2 Malfunction Tests - Malfunctions available on the simulator have been tested for certification. These tests will be scheduled for continuing testing such that 25% are tested each year and all are tested during the four year period following the submittal of this report. As malfunctions are developed/implemented, the number of certification tests will be increased to incorporate these additional malfunctions into the certification test program and included in the 25% per year schedule.

As more plant data becomes available, the simulator staff will develop additional certification tests which will be used to verify simulator fidelity. It is planned that these tests will be added to the four year test plan and included in the next certification submittal.

Appendix B lists the malfunctions which are currently certified and the schedule for testing them over the next four years. The malfunction tests are divided in such a manner that each plant system is tested each year.

4.2 Simulator Test Procedures Satisfaction of ANSI/ANS-3.5 Requirements

The certification test program covers the requirements listed in ANSI/ANS-3.5. Appendix C is a cross-reference listing which shows which tests are applicable to which ANSI/ANS-3.5 items.

4.3 Certification Test Acceptance Criteria

4.3.1 Simulator Real Time Test

For the Simulator Real Time Test, the simulation must be proceeding in real time. This test ensures that processor utilization does not exceed 100% and that the operator is not distracted by hesitation in simulator real time performance. Current plans for 1991 call for upgrading the simulator computer systems to improved performance in this area to preclude future problems as the scope of simulation is enhanced. This upgrade may be delayed until 1992 to facilitate completion of higher priority work; however, it needs to be carried out prior to or coincident with the RCS model upgrade.

4.3.2 Simulator Steady State Stability Test

For the Steady State Stability Test, the selected variables must not vary from their initial values by more than +/- 2% over a 60 minute period.

4.3.3 Simulator Steady State Accuracy Tests

For the Steady State Accuracy Tests, the simulator values for critical parameters must agree with the reference plant parameter value within +/- 2% band of the associated instrument loop range. Where available, associated instrument loop error may be added to this band. For non-critical parameters, the agreement must be within +/- 10% plus instrument error. In addition, any deviations between the simulator and the plant values are evaluated to ensure they do not detract from training. Parameters which exceed the allowable error shall result in a TR being written. All certification test TRs will be reviewed by the Simulator Review Group for their impact on training. A TR which is determined by the SRG to have a negative impact on training shall be cause to fail the associated certification test. The qualifications and makeup of the Simulator Review Group are described in Appendix D.

4.3.4 Normal Operations, Transient and Malfunction Tests

For Normal Operations, Transient, and Malfunction Tests, the parameters observed must respond in the direction assumed by a best estimate and not violate the physical laws of nature. Alarms and automatic actions which are expected to occur should occur, and other alarms and automatic actions should not occur. Certification tests are reviewed by a combination of licensed operators and engineers with extensive simulator experience. Where a test result is erroneous, a simulator TR will be written. Certification test TRs will be reviewed by the Simulator Review Group for impact on training. A TR which is determined by the SRG to have a negative impact on training shall be cause to fail the associated certification test. The Qualifications of the team which reviewed the tests is included in Appendix D.

4.4 Certification Test Abstracts

Abstracts of all certification tests are included as Appendix F to this report.

4.5 Summary of Certification Deficiencies by System

A listing of certification deficiencies by system is included as Appendix E. A discussion of each deficiency is included with the associated test abstract.

APPENDIX A

SCHEDULE OF ANNUAL OPERABILITY TESTS

Real Time Test

RTT-001	Computer Real Time Test

Simulator Physical Fidelity Test

FT-001	Simulator Physical Fidelity Test

Steady State Tests

SST-001	100 Percent Power Accuracy Test
SST-002	75 Percent Power Accuracy Test
SST-003	30 Percent Power Accuracy Test

Normal Operations Tests

NOT-001	Plant Shutdown--GP-006
NOT-002	Plant Cooldown--GP-007
NOT-003	Plant Heatup--GP-002
NOT-004	Reactor Startup--GP-004
NOT-005	Plant Startup--GP-005
NOT-006	Recovery to Rated Power Following Reactor Trip

Normal Operator Surveillance Tests

NOST-001	Power Range Heat Balance
NOST-002	CVCS/SI System Operability
NOST-003	RHR Pump Operability
NOST-004	Containment Spray Operability
NOST-005	1A-SA Emergency Diesel Generator Operability
NOST-006	Deleted
NOST-007	Main Steam Isolation Valve and Main Feedwater Isolation Valve Operability Test
NOST-008	Daily Surveillance Requirements Modes 1 and 2
NOST-009	Daily Surveillance Requirements Modes 3 and 4
NOST-010	Reactor Coolant System Leakage Evaluation
NOST-011	Shutdown Margin Calculation
NOST-012	Calculation of Quadrant Power Tilt Ratio

Normal Operator Surveillance Tests (cont.)

NOST-013	Main Steam Isolation Valve Operability Test
NOST-014	Permissives P-6 and P-10 Verification
NOST-015	1B-SB Emergency Diesel Generator Operability
NOST-016	Turbine Mechanical Overspeed Trip Test
NOST-017	AFW Pump 1B-SB Operability Test - Quarterly
NOST-018	RHR Pump 1B-SB Operability
NOST-019	Reactor Coolant Pump Seals Controlled Leakage Evaluation
NOST-020	AFW Pump 1A-SA Operability Test - Quarterly
NOST-021	CCW System Operability - Quarterly
NOST-022	Deleted
NOST-023	Control Rod and Rod Position Indication Exercise

Transient Tests

TT-001	Manual Reactor Trip
TT-002	Simultaneous Trip of all Feedwater Pumps
TT-003	Simultaneous Closure of All Main Steam Isolation Valves
TT-004	Simultaneous Trip of All Reactor Coolant Pumps
TT-005	One Reactor Coolant Pump Trip
TT-006	Turbine Trip Below P-10
TT-007	Maximum Rate Power Ramp
TT-008	Maximum Size RCS Leak Inside Containment With Loss of Off-site Power
TT-009	Maximum Size Steam Leak Inside Containment
TT-010	Slow RCS Depressurization to Saturation Using PORV's and No SI

APPENDIX B

SCHEDULE OF MALFUNCTION TESTS

MALFUNCTION TESTS FIRST YEAR

MT-10162	Failure of Rod Blocks to Block (C-2, C-3, C-4)
MT-10165	Failure of Rod Block to Block (C-5)
MT-102	Inadvertent Containment Isolation Phase B
MT-1031	Safety Injection Failure (Train B, Inadvertent)
MT-1042	Reactor Trip Breakers Fail (B fails to open)
MT-111A	Pressurizer Steam Space Leak
MT-113	Loss of Instrument Air to the Containment Building (AIR-1,1)
MT-1131	Pressurizer Relief Valve Failure (445A With P-11 Interlock)
MT-1152	Pressurizer Pressure Channel Failure (PT-445 Low)
MT-12	NSW Pump Trip and Loss of NSW
MT-1210	RCP A, B, C High Vibration
MT-1216	RCS Flow Transmitter Failure (FT-436 w)
MT-1221	Steam Generator Tube Leak (S/G B)
MT-1222	Steam Generator Tube Rupture (S/G A)
MT-1231	RCP Trip From 100 Percent Power (RCP-C)
MT-126	RCS Boron Dilution
MT-135	RHR Bypass Line Leak (Train A)
MT-136	RHR Sump Valves Fail to Open
MT-152	Turbine Protection Trip Failure
MT-22	Loss of CCW to RHR Heat Exchanger
MT-25	Letdown Heat Exchanger Tube Leak
MT-42	Logic Cabinet Urgent Failure
MT-431	Dropped Rod (One Rod)
MT-44	Stuck Rod
MT-51	Letdown Isolation Valve Failure (1CS-11)
MT-5182	Seal Injection Flow Control Valve Failure (HC-186 Closed)
MT-513	RCP Number 2 Seal Failure (RCP A)
MT-523	High Temperature Divert Valve (TCV-143) Failure
MT-526	Boric Acid Pump Trip
MT-53	Letdown Line Leak Inside Containment
MT-61	Station Blackout
MT-612	Generator Output Breakers Fail to Trip

MT-62	Loss of 120-VAC Uninterruptible Power (Power Supply SIII)
MT-63	Loss of 125-VDC Emergency Bus (DP 1B-SB)
MT-710	Condensate Pump Trip (Pump A)
MT-712	Failure of Excess Condensate Dump Valve (Closed)
MT-724	Feedline Break Outside Containment
MT-772	Feedwater Bypass Valve Failure (Open)
MT-78	Turbine Driven Auxiliary Feedwater Flow Control Valve Failure (Closed)
MT-814	Steam Failure to TDAFW Pump (1MS-72 Closed)
MT-85	Steam Generator Pressure Transmitter Failure (PT-485 High)
MT-86	Steam Generator Relief Valve Failure (Open)
MT-92	Source Range Pulse Height Discriminator Failure
MT-93	Failure of Source Range High Voltage to Disconnect
MT-96	Intermediate Range Channel Gamma Compensation Failure
MT-97	Power Range Channel Detector Failure (Low)

MALFUNCTION TESTS SECOND YEAR

MT-101	Inadvertent Containment Isolation Phase A (Train A)
MT-1032	Safety Injection Failure (Train A, Fail to Initiate)
MT-1041	Reactor Trip Breakers Fail (Both Inadvertent Open)
MT-106	False Containment Spray Actuation
MT-112	Loss of Instrument Air to the Reactor (Reactor Auxiliary Building)
MT-1132	Pressurizer Relief Valve Failure (444B Without P-11 Interlock)
MT-114	Pressurizer Safety Valve Failure (8010C Open)
MT-1151	Pressurizer Pressure Channel Failure (PT-444 High)
MT-1211	RCS Leak Within Capacity of Charging Pumps
MT-1212	LOCA Within Capacity of the SI Pumps
MT-1214	RCP Bearing Oil Reservoir Leak
MT-1215	RCS Thermal Barrier Leak into CCW System
MT-1232	Reactor Coolant Pump Trips (RCP-C)
MT-1281	RCS Protection RTD Failure (TE-412B Low)
MT-131	RHR Pump Trip (Pump A)
MT-1321	RHR HX Flow Control Valve Failure (FCV-603A Closed)
MT-151	Inadvertent Turbine Trip
MT-17	Refueling Water Storage Tank Leak
MT-210	Seal Water Heat Exchanger Tube Leak
MT-24	Component Cooling Water Header Supply Valve Failure (Closed)
MT-26	Loss of CCW to RCP Thermal Barrier
MT-31	Circulating Water Pump Trip
MT-35	Loss of Condenser Vacuum Pump
MT-41	Power Cabinet Urgent Failure
MT-45	Ejected Rod
MT-46	Uncontrolled Rod Motion
MT-47	Failure of Auto Rod Blocks to Block (C-11)
MT-514	RCP Number 3 Seal Failure (RCP C)
MT-5151	Boric Acid Flow Transmitter Failure (FT-113 to 20 gpm)
MT-5284	Charging Line Leak in Containment Before Regen HX
MT-5285	Charging Line Leak Between Regen HX and 1CS-492
MT-54	Letdown Line Leak Outside Containment
MT-55	Charging Pump Trip
MT-56	Reactor Makeup Water Pump Trip
MT-571	Letdown Pressure Control Valve Failure (PK-145 Open)
MT-65	Loss of 6.9-KV Emergency Bus (1A-SA)

MT-66	Loss of a 125-VDC Nonvital Bus (DP 1A)
MT-67	Diesel Generator Failure
MT-712A	Turbine Driven Auxiliary Feedwater Pump Trip
MT-719	Main Feedwater Pump Trip (Pump B)
MT-72	Condensate Booster Pump Trip (Pump B)
MT-720	Main Feedwater Pump Recirc Valve Failure (Pump 1B)
MT-723	Feedline Break Inside Containment
MT-73	Turbine Driven Auxiliary Feedwater Pump Speed Control Oscillates
MT-82	Steam Break Outside Containment
MT-84	Steam-Line Flow Transmitter FT-494
MT-94	Source Range Channel High Voltage Failure
MT-95	Intermediate Range Channel Failure
MT-98	Power Range Channel Failure (Low)

MALFUNCTION TESTS THIRD YEAR

MT-1013	Inadvertent Feedwater Isolation
MT-1072	Turbine Runback Failure (Failure to Runback)
MT-1110	Pressurizer Level Control Band Shift Down
MT-1162	Pressurizer Pressure Channel Failure (PT-457 Low)
MT-1171	Pressurizer Level Channel Failure (LT-459 Low)
MT-124	Reactor Coolant Pump Trip (Locked Rotor)
MT-1282	RCS Protection RTD Failure (TE-422B High)
MT-129	RCS WR Pressure Transmitter Failure (PT-403 High)
MT-13	Containment Fan Cooler Unit Trip
MT-1322	RHR HX Flow Control Valve Failure (FCV-603B Open)
MT-1331	RHR HX Bypass FCV Failure (FK-605A1 Open)
MT-137	Containment Spray Pump Failure
MT-138	Containment Spray Pump Discharge Valve Failure
MT-154	Turbine Vibration
MT-271	Letdown Temperature Controller Failure (TK-144 Low)
MT-32	Main Condenser Tube Leak
MT-333	Hotwell Level Controller Failure (LC-1901 Low)
MT-34	Loss of Condenser Vacuum
MT-410	DRPI--Open or Shorted Coil
MT-411	Improper Bank Overlap
MT-412	Control Bank Rod Step Counter Failure
MT-5111	VCT Level Transmitter Failure (LT-112 High)
MT-5201	Failure of Charging Flow Control Valve
MT-5282	Charging Pump Discharge Line Leak Before FT-122
MT-5283	Charging Line Leak Between FT-122 and 1CS-235
MT-572	Letdown Pressure Control Valve Failure (PK-145 Closed)
MT-58	Loss of Normal Letdown
MT-59	VCT Divert Valve Control Failure (HUT)
MT-610	Loss of Unit Auxiliary Transformer
MT-64	Loss of 6.9 KV Auxiliary Bus (1B)
MT-68	Automatic Voltage Regulator Failure (High)
MT-69	Loss of Start-up Transformer
MT-711A	Motor Driven Auxiliary Feedwater Pump Trip
MT-721	Feedwater Flow Transmitter Failure (FT-466 Low)
MT-722	Feedwater Control Valve Position Failure (LCV-488 Open)
MT-725	Steam Generator Level Channel Failure (LT-496 Low)

MT-76	Auxiliary Feedwater Flow Control Valve Failure (Open)
MT-81	Steamline Break Inside Containment
MT-815	Main Steam Header Break
MT-83	Steam Header Pressure Detector Failure (PT-464 High)
MT-87	MSIV Failure (S/G B Shut)
MT-89	Atmospheric Steam Dump Valve Failure (PCV-408J Open)
MT-91	Source Range Instrument Failure (N31 High)

MALFUNCTION TESTS FOURTH YEAR

MT-1014	Inadvertent Main Steam Isolation
MT-1015	Diesel Generator Sequencer Fails to Complete Block 1
MT-10161	Failure of Rod Blocks to Block (C-1)
MT-1017	Failure of Permissive Interlock P-14
MT-1071	Turbine Runback Failure (Erroneous Runback)
MT-111	Loss of Instrument Air (Turbine Building)
MT-112A	Pressurizer Spray Valve Failure
MT-1161	Pressurizer Pressure Channel Failure (PT-456 High)
MT-1172	Pressurizer Level Channel Failure (LT-459 High)
MT-118	Pressurizer Backup Heaters Groups A and B Failure
MT-1211A	RCS Fuel Rod Breach
MT-1212A	RCS Leakage into an Accumulator
MT-1213	RCS Vessel Flange Leak
MT-125	RCP Shaft Break Accident (RCP B)
MT-1271	RC Control RTD Failure (TE-411B High)
MT-1332	RHR HX Bypass FCV Failure (FK-605B1 Closed)
MT-134	RHR to Letdown Valve Failure (HCV-142.1 Open)
MT-1519	Turbine First Stage Pressure Transmitter Failure (PT-446 Low)
MT-155	Governor Valve Failure (GV-3 Closed)
MT-157	Turbine DEH Computer Failure
MT-21	Component Cooling Water Pump Trip
MT-23	CCW Leak into the Service Water System
MT-272	Letdown Temperature Controller Failure (TK-144 High)
MT-28	Loss of CCW to the Reactor Coolant Pumps
MT-331	Hotwell Level Controller Failure (LC-1900 High)
MT-413	Rod Speed Deadband Control Failure
MT-48	TREF Failure
MT-49	DRPI--Loss of Voltage
MT-5112	VCT Level Transmitter Failure (LT-115 Low)
MT-512	RCP Number 1 Seal Failure (RCP B)
MT-5152	Boric Acid Flow Transmitter Failure (FT-113 to 0 gpm)
MT-516	Boric Acid Filter Plugged
MT-5181	Seal Injection Flow Control Valve Failure (HC-186 Open)
MT-5202	Failure of Charging Flow Control Valve (Closed)
MT-524	Charging Pump Suction From RWST Failure (115D Open)
MT-525	Charging Pump Mini Flow Valve Failure (1CS-182 Closed)

MT-527	Charging Line Containment Isolation Valve Failure
MT-5281	Charging Line Leak on Charging Pump Suction
MT-615	Diesel Generator Governor Failure
MT-616	Diesel Generator Breaker Inadvertent Trip
MT-714	Condensate Storage Tank Leak
MT-715	Heater Drain Pump Trip (Pump B)
MT-771	Feedwater Bypass Valve Failure (Closed)
MT-810	Steam Dump Control Failure (Closed)
MT-811	Mechanically Stuck Condenser Dump Valve (PCV-408 Open)
MT-812	Steam Dump Permissive (P-12) Failure
MT-88	Steam Generator Safety Valve Failure (Open)
MT-911	Source Range Instrument Power Fuse Blown
MT-912	Intermediate Range Control Power Fuse Blown
MT-913	Power Range Control Power Fuse Blown

APPENDIX C

CERTIFICATION TEST TO ANSI/ANS-3.5 CROSS-REFERENCE

I. COMPUTER REAL TIME

Appendix A, Section 3.1

RTT-001	Computer Real-Time Test

II. STEADY STATE OPERATION

Section 4.1 and Appendix B2.1

SST-001	Steady State Accuracy and Stability--100 Percent Power
SST-002	Steady State Accuracy--75 Percent Power
SST-003	Steady State Accuracy--30 Percent Power

III. NORMAL PLANT EVOLUTIONS

Section 3.1.1 (1)

NOT-003	Plant Heat-up--GP-002

Section 3.1.1 (2)

NOT-004	Reactor Start-up--GP-004
NOT-005	Plant Start-up--GP-005

Section 3.1.1 (3)

NOT-005	Plant Start-up--GP-005

Section 3.1.1 (4)

NOT-006	Recovery to Rated Power Following Reactor Trip

Section 3.1.1 (5)

This section is satisfied by General Procedures GP-004, GP-006 and GP-007 which either begin from Hot Standby conditions or end with the plant in hot standby. There are no procedures written which specifically address operations at hot standby.

Section 3.1.1 (6)

NOT-005	Plant Start-up--GP-005

III. NORMAL PLANT EVOLUTIONS (cont.)

Section 3.1.1 (7)

This section is not applicable. Power operations with less than three operating reactor coolant pumps is prohibited by Technical Specifications.

Section 3.1.1 (8)

NOT-001	Plant Shutdown--GP-006
NOT-002	Plant C/D--GP-007

Section 3.1.1 (9)

NOST-001	OST-1004	Power Range Heat Balance
NOST-011	OST-1036	Shutdown Margin Calculation
NOST-012	OST-1039	Calculation of Quadrant Power Tilt Ratio
NOT-006	N/A	Estimated Critical Condition Calculation performed with NOT-006

Section 3.1.1 (10)

NOST-001	OST-1004	Power Range Heat Balance
NOST-002	OST-1007	CVCS/SI System Operability
NOST-003	OST-1008	RHR Pump Operability
NOST-004	OST-1009	Containment Spray Operability
NOST-005	OST-1013	1A-SA Emergency Diesel Generator Operability
NOST-007	OST-1018	Main Steam Isolation Valve and Main Feed-water Isolation Valve Operability Test
NOST-008	OST-1021	Daily Surveillance Requirements Modes 1 and 2
NOST-009	OST-1022	Daily Surveillance Requirements Modes 3 and 4
NOST-010	OST-1026	Reactor Coolant System Leakage Evaluation
NOST-011	OST-1036	Shutdown Margin Calculation
NOST-012	OST-1039	Calculation of Quadrant Power Tilt Ratio
NOST-013	OST-1046	Main Steam Isolation Valve Operability Test
NOST-014	OST-1054	Permissives P-6 and P-10 Verification
NOST-015	OST-1073	1B-SB Emergency Diesel Generator Operability
NOST-016	OST-1075	Turbine Mechanical Overspeed Trip Test
NOST-017	OST-1076	AFW Pump 1B-SB Operability Test - Quarterly
NOST-018	OST-1092	RHR Pump 1B-SB Operability
NOST-019	OST-1126	Reactor Coolant Pump Seals Controlled Leakage Evaluation
NOST-020	OST-1211	AFW Pump 1A-SA Operability Test - Quarterly

III. NORMAL PLANT EVOLUTIONS (cont.)

NOST-021	OST-1316	CCW System Operability--Quarterly
NOST-023	OST-1005	Control Rod and Rod Position Indication Exercise

IV. TRANSIENTS

Section 4.2 and Appendix B.2.2

TT-001	Manual Reactor Trip
TT-002	Simultaneous Trip of All Feedwater Pumps
TT-003	Simultaneous Closure of All Main Steam Isolation Valves
TT-004	Simultaneous Trip of All Reactor Coolant Pumps
TT-005	One Reactor Coolant Pump Trip
TT-006	Turbine Trip Below P-10
TT-007	Maximum Rate Power Ramp
TT-008	Maximum Size RCS Leak Inside Containment With Loss of Off-site Power
TT-009	Maximum Size Steam Leak Inside Containment
TT-010	Slow RCS Depressurization to Saturation Using Pzr PORVs and No SI

V. MALFUNCTIONS

Section 3.1.2(1)(a)

MT-1221	Steam Generator Tube Leak (S/G B)
MT-1222	Steam Generator Tube Rupture (S/G A)

Section 3.1.2(1)(b)

MT-53	Letdown Line Leak Inside Containment
MT-54	Letdown Line Leak Outside Containment
MT-5281	Charging Line Leak on Charging Pump Suction
MT-5282	Charging Pump Discharge Line Leak Before FT-122
MT-5283	Charging Line Leak Between FT-122 and 1CS-235
MT-5284	Charging Line Leak in Containment Before Regen HX
MT-5285	Charging Line Leak Between Regen HX and 1CS-492
MT-111A	Pressurizer Steam Space Leak
MT-25	Letdown Heat Exchanger Tube Leak
MT-135	RHR Bypass Line Leak (Train A)
MT-1215	RCS Thermal Barrier Leak into CCW System

V. **MALFUNCTIONS** (cont.)

Section 3.1.2(1)(c)

MT-111A	Pressurizer Steam Space Leak
MT-1211	RCS Leak Within Capacity of Charging Pumps
MT-1212	LOCA Within Capacity of the SI Pumps
MT-1213	RCS Vessel Flange Leak

Section 3.1.2(1)(d)

MT-86	Steam Generator Relief Valve Failure (Open)
MT-88	Steam Generator Safety Valve Failure (Open)
MT-89	Atmospheric Steam Dump Valve Failure (PCV-408J Open)
MT-1131	Pressurizer Relief Valve Failure (445A With P-11 Interlock)
MT-1132	Pressurizer Relief Valve Failure (444B Without P-11 Interlock)
MT-114	Pressurizer Safety Valve Failure (8010C Open)

Section 3.1.2(2)

MT-111	Loss of Instrument Air (Turbine Bldg.)
MT-112	Loss of Instrument Air to the Reactor Auxiliary Building
MT-113	Loss of Instrument Air to the Containment Building

Section 3.1.2(3)

MT-49	DRPI--Loss of Voltage
MT-61	Station Blackout
MT-62	Loss of 120 VAC Uninterruptible Power Supply (SIII)
MT-64	Loss of 6.9-kV Auxiliary Bus (1B)
MT-65	Loss of 6.9-kV Emergency Bus (1A-SA)
MT-67	Diesel Generator Failure
MT-68	Automatic Voltage Regulator Failure (High)
MT-69	Loss of Start-up Transformer
MT-610	Loss of Unit Auxiliary Transformer
MT-615	Diesel Generator Governor Failure
MT-616	Diesel Generator Breaker Inadvertent Trip
MT-1015	Diesel Generator Fails to Complete Block One
MT-63	Loss of 125 VDC Emergency Bus (DP 1B-SB)
MT-66	Loss of 125 VDC Nonvital Bus (DP 1A)

V. MALFUNCTIONS (cont.)

Section 3.1.2(4)

MT-1231	RCP Trip From 100 Percent Power (RCP-C)
MT-1232	Reactor Coolant Pump Trip (RCP-C)
MT-124	Reactor Coolant Pump Trip (Locked Rotor)
MT-125	RCP Shaft Break Accident (RCP B)

Section 3.1.2(5)

MT-31	Circulating Water Pump Trip
MT-331	Hotwell Level Controller Failure
MT-333	Hotwell Level Controller Failure (LC-1901 Low)
MT-34	Loss of Condenser Vacuum
MT-35	Loss of Condenser Vacuum Pump
MT-712	Failure of Excess Condensate Dump Valve (Closed)

Section 3.1.2(6)

MT-12	NSW Pump Trip and Loss of NSW

Section 3.1.2(7)

MT-22	Loss of CCW to RHR Heat Exchanger
MT-131	RHR Pump Trip (Pump A)
MT-1321	RHR HX Flow Control Valve Failure (FCV-603A Closed)
MT-1322	RHR HX Flow Control Valve Failure (FCV-603B Open)
MT-1331	RHR HX Bypass FCV Failure (FK-605A1 Open)
MT-1332	RHR HX Bypass FCV Failure (FK-605B1 Closed)
MT-134	RHR to Letdown Valve Failure (HCV-142.1 Open)
MT-136	RHR Sump Valves Fail to Open

Section 3.1.2(8)

MT-21	Component Cooling Water Pump Trip
MT-22	Loss of CCW to RHR Heat Exchanger
MT-23	CCW Leak Into The Service Water System
MT-24	Component Cooling Water Header Supply Valve Failure (Closed)
MT-26	Loss of CCW to RCP Thermal Barrier
MT-28	Loss of CCW To The Reactor Coolant Pumps

Section 3.1.2(9)

MT-72	Condensate Booster Pump Trip (Pump B)
MT-771	Feedwater Bypass Valve Failure (Closed)
MT-772	Feedwater Bypass Valve Failure (Open)
MT-710	Condensate Pump Trip (Pump A)
MT-714	Condensate Storage Tank Leak
MT-715	Heater Drain Pump Trip (Pump B)
MT-719	Main Feedwater Pump Trip (Pump B)
MT-720	Main Feedwater Pump Recirc Valve Failure (Pump 1B)
MT-721	Feedwater Flow Transmitter Failure (FT-466 Low)
MT-722	Feedwater Control Valve Position Failure (LCV-488 Open)

Section 3.1.2(10)

MT-1013	Inadvertent Feedwater Isolation
MT-711A	Motor Driven Auxiliary Feedwater Pump Trip
MT-712A	Turbine Driven Auxiliary Feedwater Pump Trip

Section 3.1.2(11)

MT-725	Steam Generator Level Channel Failure (LT-496 Low)
MT-1161	Pressurizer Pressure Channel Failure (PT-456 High)
MT-1162	Pressurizer Pressure Channel Failure (PT-457 Low)
MT-1281	RCS Protection RTD Failure (TE-412B Low)
MT-1282	RCS Protection RTD Failure (TE-422B High)

Section 3.1.2(12)

MT-41	Power Cabinet Urgent Failure
MT-431	Dropped Rod (One Rod)
MT-44	Stuck Rod
MT-45	Ejected Rod
MT-46	Uncontrolled Rod Motion
MT-47	Failure of Auto Rod Blocks to Block
MT-48	TREF Failure
MT-411	Improper Bank Overlap
MT-412	Control Bank Rod Step Counter Failure
MT-413	Rod Speed Deadband Control failure

V. **MALFUNCTIONS** (cont.)

Section 3.1.2(13)

MT-42	Logic Cabinet Urgent Failure

Section 3.1.2(14)

MT-1211A	RCS Fuel Rod Breach

Section 3.1.2(15)

MT-151	Inadvertent Turbine Trip

Section 3.1.2(16)

MT-610	Loss of Unit Auxiliary Transformer

Section 3.1.2(17)

MT-5151	Boric Acid Flow Transmitter Failure (FT-113 to 20 GPM)
MT-5152	Boric Acid Flow Transmitter Failure (FT-113 to 0 GPM)
MT-516	Boric Acid Filter Plugged
MT-526	Boric Acid Pump Trip
MT-810	Steam Dump Control Failure (closed)
MT-811	Mechanically Stuck Condenser Dump Valve (PCV-408A Open)
MT-812	Steam Dump Permissive (P-12) Failure
MT-1013	Inadvertent Feedwater Isolation
MT-1014	Inadvertent Main Steam Isolation

Section 3.1.2(18)

MT-1161	Pressurizer Pressure Channel Failure (PT-456) high
MT-1162	Pressurizer Pressure Channel Failure (PT-457) low
MT-1131	Pressurizer Relief Valve Failure (445A open with P-11)
MT-1132	Pressurizer Relief Valve Failure (444B open without P-11)
MT-51	Letdown Isolation Valve Failure (1CS-11)
MT-53	Letdown Line Leak Inside Containment
MT-54	Letdown Line Leak Outside Containment
MT-55	Charging Pump Trip
MT-56	Reactor Makeup Water Pump Trip
MT-571	Letdown Pressure Control Valve Failure (PK-145 Open)
MT-572	Letdown Pressure Control Valve Failure (PK-145 Closed)
MT-58	Loss of Normal Letdown

V. MALFUNCTIONS (cont.)

MT-59	VCT Divert Valve Control Failure (HUT)
MT-5111	VCT Level Transmitter Failure (LT-112 High)
MT-5112	VCT Level Transmitter Failure (LT-115 Low)
MT-512	RCP Number 1 Seal Failure (RCP B)
MT-513	RCP Number 2 Seal Failure (RCP A)
MT-514	RCP Number 3 Seal Failure (RCP C)
MT-112A	Pressurizer Spray Valve Failure
MT-5151	Boric Acid Flow Transmitter Failure (FT-113 to 20 GPM)
MT-5152	Boric Acid Flow Transmitter Failure (FT-113 to 0 GPM)
MT-516	Boric Acid Filter Plugged
MT-5181	Seal Injection Flow Control Valve Failure (HC-186 Open)
MT-5182	Seal injection Flow Control Valve Failure (HC-186 Closed)
MT-5201	Failure of Charging Flow Control Valve
MT-5202	Failure of Charging Flow Control Valve (Closed)
MT-523	High Temperature Divert Valve (TCV-143) Failure
MT-524	Charging Pump Suction from RWST Failure (115D Open)
MT-525	Charging Pump Mini Flow Valve Failure (1CS-182 Closed)
MT-526	Boric Acid Pump Trip
MT-527	Charging Line Containment Isolation Valve Failure
MT-5281	Charging Line Leak on Charging Pump Suction
MT-5282	Charging Pump Discharge Line Leak Before FT-122
MT-5283	Charging Line Leak Between FT-122 and 1CS-235
MT-5284	Charging Line Leak in Containment Before Regen HX.
MT-5285	Charging Line Leak Between Regen HX and 1CS-492

Section 3.1.2(19)

MT-1041	Reactor Trip Breakers Fail (Both Inadvertent Open)

Section 3.1.2(20)

MT-723	Feedline Break Inside Containment
MT-724	Feedline Break Outside Containment
MT-81	Steamline Break Inside Containment
MT-82	Steamline Break Outside Containment
MT-815	Main Steam Header Break

V. MALFUNCTIONS (cont.)

Section 3.1.2(21)

MT-91	Source Range Instrument Failure (N31 High)
MT-92	Source Range Pulse Height Discriminator Failure
MT-93	Failure of Source Range High Voltage to Disconnect
MT-94	Source Range Channel High Voltage Failure
MT-95	Intermediate Range Channel Failure
MT-96	Intermediate Range Channel Gamma Compensation Failure
MT-97	Power Range Channel Detector Failure (Low)
MT-98	Power Range Channel Failure (Low)
MT-911	Source Range Instrument Power Fuse Blown
MT-912	Intermediate Range Control Power Fuse Blown
MT-913	Power Range Control Power Fuse Blown

Section 3.1.2(22)

MT-271	Letdown Temperature Controller Failure (TK-144 Low)
MT-272	Letdown Temperature Controller Failure (TK-144 High)
MT-331	Hotwell Level Controller Failure (LC-1900 High)
MT-333	Hotwell Level Controller Failure (LC-1901 Low)
MT-47	Failure of Auto Rod Blocks to Block (C-11)
MT-48	TREF Failure
MT-5111	VCT Level Transmitter Failure (LT-112 High)
MT-5112	VCT Level Transmitter Failure (LT-115 Low)
MT-5151	Boric Acid Flow Transmitter Failure (FT-113 to 20 GPM)
MT-5152	Boric Acid Flow Transmitter Failure (FT-113 to 0 GPM)
MT-721	Feedwater Flow Transmitter Failure (FT-466 Low)
MT-722	Feedwater Control Valve Position Failure (LCV-488 Open)
MT-725	Steam Generator Level Channel Failure (LT-496 Low)
MT-83	Steam Header Pressure Detector Failure (PT-464 High)
MT-84	Steam-Line Flow Transmitter FT-494 Failure (Low)
MT-85	Steam Generator Pressure Transmitter Failure (PT-485 High)
MT-810	Steam Dump Control Failure (Closed)
MT-812	Steam Dump Permissive (P-12) Failure
MT-101	Inadvertent Containment Isolation Phase A (Train A)
MT-102	Inadvertent Containment Isolation Phase B
MT-1031	Safety Injection Failure (Train B, Inadvertent)
MT-1032	Safety Injection Failure (Train A, Fail to Initiate)

V. <u>MALFUNCTIONS</u> (cont.)

MT-106	False Containment Spray Actuation
MT-1071	Turbine Runback Failure (Erroneous Runback)
MT-1072	Turbine Runback Failure (Failure to Runback)
MT-1013	Inadvertent Feedwater Isolation
MT-1014	Inadvertent Main Steam Isolation
MT-10161	Failure of Rod Blocks to Block (C-1)
MT-10162	Failure of Rod Blocks to Block (C-2, C-3, C-4)
MT-10165	Failure of Rod Blocks to Block (C-5)
MT-1017	Failure of Permissive Interlock P-14
MT-1131	Pressurizer Relief Valve Failure (445A Open With P-11)
MT-1132	Pressurizer Relief Valve Failure (444B Open Without P-11)
MT-1151	Pressurizer Pressure Channel Failure (PT-444 High)
MT-1152	Pressurizer Pressure Channel Failure (PT-445 Low)
MT-1161	Pressurizer Pressure Channel Failure (PT-456 High)
MT-1162	Pressurizer Pressure Channel Failure (PT-457 Low)
MT-1171	Pressurizer Level Channel Failure (LT-459 Low)
MT-1172	Pressurizer Level Channel Failure (LT-459 High)
MT-118	Pressurizer Backup Heaters Groups A and B Failure
MT-1110	Pressurizer Level Control Band Shift Down
MT-1271	RCS Control RTD Failure (TE-411B High)
MT-1281	RCS Protection RTD Failure (TE-412B Low)
MT-1282	RCS Protection RTD Failure (TE-422B High)
MT-129	RCS WR Pressure Transmitter Failure (PT-403 High)
MT-1216	RCS Flow Transmitter Failure (FT-436 Low)
MT-1332	RHR HX Bypass FCV Failure (FK-605B1 Closed)
MT-1519	Turbine First-Stage Pressure Transmitter Failure (PT-446 Low)

Section 3.1.2(23)

MT-524	Charging Pump Suction from RWST Failure (115D Open)
MT-525	Charging Pump Mini Flow Valve Failure (8109B Closed)
MT-527	Charging Line Containment Isolation Valve Failure
MT-67	Diesel Generator Failure
MT-711A	Auxiliary Feedwater Pump Trip (Pump A)
MT-712A	Auxiliary Feedwater Pump Trip (Turbine)
MT-73	Turbine Driven Auxiliary Feedwater Pump Speed Control Oscillates

V. MALFUNCTIONS (cont.)

MT-76	Auxiliary Feedwater Flow Control Valve Failure (Open)
MT-78	Turbine Driven Auxiliary Feedwater Flow Control Valve Failure (Closed)
MT-814	Steam Failure to TDAFW Pump (1MS-72 Closed)
MT-1212A	RCS Leakage into an Accumulator
MT-136	RHR Sump Valves Fail to Open
MT-138	Containment Spray Pump Discharge Valve Failure
MT-17	Refueling Water Storage Tank Leak

Section 3.1.2(24)

MT-1042	Reactor Trip Breakers Fail (B fails to open)

VI. SIMULATOR ENVIRONMENT

Section 3.2

FT-001	Simulator Physical Fidelity Test

APPENDIX D

DESCRIPTION AND QUALIFICATIONS OF SIMULATOR REVIEW GROUP AND SIMULATOR TEST REVIEW TEAM

The Simulator Review Group (SRG) was organized to ensure the validity and sufficiency of the simulator configuration. The group reviews several items to insure the scope and accuracy of simulation is maintained adequately for training and to meet the guidelines of ANSI/ANS-3.5. Specifically, the group reviews plant changes which affect the simulator but which, due to training value, may not be completely implemented in the simulator. The group also reviews hardware and software discrepancies which are outstanding in the simulator to insure that training can continue without negative effects. Minutes are kept of SRG meetings and they are filed with the simulator records.

The SRG is comprised of the Harris Plant Manager - Operations or designee from operations management; a currently licensed and active control room reactor or senior reactor operator; the Manager - Licensed Training or designee; an SRO licensed or certified simulator instructor; Manager - Simulator (SRG Chairman); and a simulator operations specialist who is or has been SRO licensed or certified. At least 3 members must be present, one from each group, for the SRG to convene its meetings. The SRG is required to meet at least once per calendar year, is scheduled to meet approximately every 3 months, and has met about once every 4-5 weeks since October 1990. The primary members (designated by an asterisk--*) or their designees/alternates, along with their qualifications, are as follows:

John Boska* -

> BA degree in Physics, MS in Electrical Engineering. Five years Navy nuclear operating experience. Five years as SRO certified instructor at Westinghouse SNUPPS simulator. Three years as licensed class instructor at Harris Nuclear Plant. One year as Manager - Licensed Training at Harris Nuclear Plant.

J. T. Bryan* -

BS degree in Electrical Engineering. MS degree in Computer Science. Registered Professional Engineer in North Carolina. Thirteen years experience in real time simulator software development and management.

Keith Holbrook -

Six years of Navy Nuclear operating experience. Five years on shift at auxiliary and licensed Reactor Operator at Harris Nuclear Plant.

Ron Leblond* -

Nine years as a Navy Reactor Operator including 3 years of prototype training experience. Certified by the NRC as a Senior Reactor Operator at the Palisades Nuclear Plant. Two and one half years experience as a simulator instructor. NRC certified as a Senior Reactor Operator at the Harris Nuclear Plant. Six years experience as a simulator instructor and a simulator operations specialist.

Mark Palmer -

Six years Navy nuclear operating experience. Senior Reactor Operator License at Harris Nuclear Plant 1984. Five years as senior reactor operator on shift at Harris Nuclear Plant including plant startup test program performance. One year as SRO on Shift Foreman.

Jim Pierce* -

Eight years of Navy Nuclear operating experience. Four years as licensed operator on shift at HB Robinson. Two years as licensed instructor at HB Robinson. Certified as SRO at Harris Nuclear Plant 1985. Six years as licensed operator instructor at Harris Nuclear Plant.

Robert Smith -

Two and one half years of licensed operator and non-licensed operator experience at Brunswick Steam Electric Plant. Four years in license class and on shift as operator at Harris Nuclear Plant. Four years as licensed operator instructor at Harris Nuclear Plant.

Dean Tibbitts* -

BS and MS Nuclear Engineering 1975. Five years as Project Engineer for USNRC. Three years as nuclear engineering consultant. Six years as Regulatory Compliance Specialist and Director - Regulatory Compliance at Harris Nuclear Plant. Six months as Shift Operating Supervisor - Nuclear at Harris Nuclear Plant.

Robert Winkler -

Six years Navy Nuclear operating experience. Two years license class and four years of on shift operating experience at Harris Nuclear Plant. One year of simulator operations support and one year of licensed training instructor at Harris Nuclear Plant.

To ensure the validity and sufficiency of the certification test program, certification tests are prepared by a licensed or certified Senior Reactor Operator and reviewed by a second SRO or engineer. The tests are performed by a licensed or certified SRO assisted, where relevant, by a senior reactor operator currently on shift at the Harris Plant. The results of simulator certification tests are reviewed by at least 3 persons (consisting of one or more engineers and one or more senior reactor operators) who have extensive simulator and/or operations experience. Primary members of the SRG along with the following persons have been involved with review of the certification tests and function as the "panel of experts" to validate test results. Their qualifications are as follows:

Ron Bright -

BS degrees in Nuclear Engineering and Electrical Engineering. Two years experience performing nuclear plant transient analysis for Babcock and Wilcox. Four years experience installing and tuning Emergency Response Facility computer system at Harris and Robinson Nuclear Plants. Three years as simulator support engineer at the Harris simulator.

Randy Houck -

BS Nuclear Engineering. Three years as nuclear plant chemical engineer. Three years of real time simulator software development.

APPENDIX E

SUMMARY OF CERTIFICATION DEFICIENCIES BY SYSTEM

Deficiencies that were discovered during the performance of the certification tests are contained in this appendix to the certification submittal package. The deficiencies are organized in alphabetical order by system and list the number of the associated test procedure. A description and impact on training of the deficiency are contained in the respective certification test abstract in Appendix F. There are currently 43 deficiencies outstanding against 22 of 230 certification tests. There are currently 2 tests rated as unsatisfactory for which there is no work-around. The deficiencies will be corrected by March 26, 1992.

There is a program in place which tracks all outstanding deficiencies and identifies a resolution schedule consistent with regulatory requirements, training impact, and training schedules. In no case will the deficiency remain unresolved more than 2 years beyond identification.

After each specific deficiency was evaluated, the test abstracts containing deficiency impact on training were presented to the Simulator Review Group (SRG) for their review and approval. By their approval, the SRG agrees that it is acceptable to continue with training and license examination while the deficiencies are being corrected.

TEST/RESULTS	SSR#	TITLE
ALARMS:		
NONE		
AUXILIARY FEEDWATER SYSTEM:		
NOT-003/S	90-0772	AFW MOTOR RECIRC FLOW
NOST-020/S	90-0772	AFW MOTOR RECIRC FLOW
NOST-017/S	90-0772	AFW MOTOR RECIRC FLOW
AXIAL POWER MONITOR (EXCORE):		
NONE		
AUXILIARY STEAM:		
NONE		

COMPRESSED AIR:

NONE

COMPONENT COOLING WATER SYSTEM:

SST-001/S	91-0074	CCW TI-671 TOO HIGH IN THE SIMULATOR

CONDENSATE AND FEEDWATER SYSTEM:

NOT-001/S	90-0760	SECONDARY SIDE MASS CONSERVATION
SST-003/S	91-0077	SIMULATOR VACUUM TOO LOW

CONTAINMENT HVAC SYSTEM:

NONE

CONTAINMENT SPRAY SYSTEM:

NONE

CONTROL ROD DRIVE AND DRPI SYSTEM:

MT-41/S	90-0267	URGENT FAILURE ON C SHUTDOWN BANK
MT-49/S	90-0270	TWO OR MORE RODS AT BOTTOM ANNUNCIATOR

COOLING WATER SYSTEM:

NONE

CHEMICAL AND VOLUME CONTROL SYSTEM:

NOT-001/S	90-0780	BTRS NOT FUNCTIONAL
NOT-002/S	90-0818	SEAL INJECT WITH CSIP OFF

CIRCULATING WATER SYSTEM:

NOT-005/S	90-0830	ERFIS POINTS ON CIRC WATER

DIGITAL ELECTROHYDRAULIC CONTROL SYSTEM:

MT-157/S	90-0862	DEH REF INDICATION FOLLOWING CONTROLLER RESET

DIESEL GENERATOR SYSTEM:

MT-616/S	90-0718	TRIP ANNUN NOT SEALED IN

DIESEL SEQUENCER SYSTEM:

NONE

ESSENTIAL SERVICES CHILL WATER:

NONE

ELECTRICAL DISTRIBUTION SYSTEM:

MT-610/S	90-0728	RELAY RESPONSE TO LOSS OF UAT

ERFIS INTERFACE SYSTEM:

MT-13/S	90-0982	ERFIS POINTS FOR CONTAINMENT FAN COOLERS
MT-1015/S	91-0022	ERFIS PTS 2EE 1751 AND 2EE 1726 ON SDS

FIRE PROTECTION SYSTEM:

NONE

FEEDWATER HEATER, EXTRACTIONS DRAINS, AND VENTS SYSTEM:

NONE

GENERATOR SYSTEM:

MT-68/S	90-0699	MS1 GEN1 DIDN'T ALARM ON VOLTS/FREQ
NOT-005/S	90-0842	MW RECORDER ER-0569
NOT-006/S	90-0842	MW RECORDER ER-0569

GROSS FAILED FUEL DETECTOR SYSTEM:

NONE

HARDWARE SYSTEM:

RTT-001/S	91-0080	MINOR PAUSES/HESITATION NOTED IN ANNUNCIATORS
NOST-008/S	90-0707	MIMS REQUIRED BY DSR
FT-001	90-0146	STATUS BOARD DOES NOT MATCH CONTROL ROOM
FT-001	90-0845	NEED CHAIN ACROSS ACCESS
FT-001	90-0917	MAIN CONTROL ROOM SOUND DETECTOR
FT-001	90-1010	SIMULATOR DEMARCATION
FT-001	90-1011	SIMULATOR METER LEGENDS AND SCALES
FT-001	90-1012	SIMULATOR BOARD LABELS

FT-001	90-1013	CSIP AUX LUBE OIL PUMP STATUS LIGHT ENGRAVINGS
FT-001	90-0133	BEEPER ON RADIO CONSOLE
FT-001	90-0134	MISSING RELAY ON RELAY PANEL
FT-001	90-0135	MISCOLORED SLBs ON AEP-1
FT-001	90-0136	RM-23 RECORDERS

<u>HVAC SYSTEM</u>:

NONE

<u>INCORE FLUX MAPPING SYSTEM</u>:

NONE

<u>REACTOR AND AUX BLDG SUMPS</u>:

NONE

<u>MISCELLANEOUS</u>:

NONE

<u>MAIN STEAM SYSTEM</u>:

NONE

<u>NUCLEAR INSTRUMENTATION SYSTEM (EXCORE)</u>:

NOST-001/S	90-0723	NI PWR LEVEL AT 100%
NOT-001/S	90-0779	NIS CHANNEL DEVIATION ALARM
NOT-003/S	90-0787	SSR AND IR RESPONSE TO HEAT UP
SST-001/S	91-0072	INTERMEDIATE RANGE TOO LOW IN SIM
SST-001/S	91-0073	POWER RANGE DELTA FLUX TOO LOW IN SIM
SST-001/S	91-0075	POWER RANGE NI CURRENTS TOO LOW IN SIM

<u>NUCLEAR SAMPLING SYSTEM</u>:

NONE

<u>PLANT CONTROL SYSTEM</u>:

NONE

<u>PLANT PROTECTION SYSTEM</u>:

NONE

PRESSURIZER SYSTEM:

NOT-003/S	90-0804	PRS RESPONSE TO LOW PRESSURE
NOT-002/S	90-0816	PZR VAPOR/LIQUID TEMP INDICATION

PRESSURE RELIEF TANK SYSTEM:

NONE

REACTOR COOLANT PUMP SYSTEM:

NONE

REACTOR COOLANT SYSTEM:

NOST-001/S	90-0711	RCP HEAT INPUT
TT-008/U	91-0066	PZR REFILLS WITH LARGE LOCA IN PROGRESS
TT-010/U	91-0067	RCS FLOW STEP DECREASE AT SATURATION

RESIDUAL HEAT REMOVAL SYSTEM:

NONE

RADIATION MONITORING INTERFACE SYSTEM:

MT-1211A/S	90-0120	RADIATION MONITOR 3501-SA LOCKED UP

REACTOR MAKEUP WATER SYSTEM:

NONE

REACTOR CORE SYSTEM:

NONE

REACTOR VESSEL LEVEL SYSTEM:

NONE

SPENT FUEL COOLING SYSTEM:

NONE

STEAM GENERATOR BLOWDOWN SYSTEM:

NONE

STEAM GENERATOR SYSTEM:

NONE

SAFETY INJECTION SYSTEM:

NONE

SWITCHYARD SYSTEM:

NONE

SERVICE WATER SYSTEM:

NONE

SYNCHRONIZER SYSTEM:

NONE

TURBINE GENERATOR AUXILIARIES SYSTEM:

NOT-005/S	90-0833	TURB ECC AND VIBRATION < 600 RPM
NOT-005/S	90-0839	TURBINE AUX COOLERS

TURBINE SUPERVISORY INSTRUMENTATION SYSTEM:

NONE

TURBINE SYSTEM:

NONE

APPENDIX F

SIMULATOR CERTIFICATION TEST ABSTRACTS

INDEX OF ABSTRACTS

Simulator Physical Fidelity Test (1)

Malfunction Tests (188)

MT-1171	Pressurizer Level Channel Failure (LT-459 Low)
MT-1172	Pressurizer Level Channel Failure (LT-459 High)
MT-118	Pressurizer Backup Heaters Groups A and B Failure
MT-12	NSW Pump Trip and Loss of NSW
MT-1210	RCP A, B, C High Vibration
MT-1211	RCS Leak Within Capacity of Charging Pumps
MT-1211A	RCS Fuel Rod Breach
MT-1212	LOCA Within Capacity of the SI Pumps
MT-1212A	RCS Leakage into an Accumulator
MT-1213	RCS Vessel Flange Leak
MT-1214	RCP Bearing Oil Reservoir Leak
MT-1215	RCS Thermal Barrier Leak into CCW System
MT-1216	RCS Flow Transmitter Failure (FT-436 w)
MT-1221	Steam Generator Tube Leak (S/G B)
MT-1222	Steam Generator Tube Rupture (S/G A)
MT-1231	RCP Trip From 100 Percent Power (RCP-C)
MT-1232	Reactor Coolant Pump Trips (RCP-C)
MT-124	Reactor Coolant Pump Trip (Locked Rotor)
MT-125	RCP Shaft Break Accident (RCP B)
MT-126	RCS Boron Dilution
MT-1271	RCS Control RTD Failure (TE-411B High)
MT-1281	RCS Protection RTD Failure (TE-412B Low)
MT-1282	RCS Protection RTD Failure (TE-422B High)
MT-129	RCS WR Pressure Transmitter Failure (PT-403 High)
MT-13	Containment Fan Cooler Unit Trip
MT-131	RHR Pump Trip (Pump A)
MT-1321	RHR HX Flow Control Valve Failure (FCV-603A Closed)
MT-1322	RHR HX Flow Control Valve Failure (FCV-603B Open)
MT-1331	RHR HX Bypass FCV Failure (FK-605A1 Open)
MT-1332	RHR HX Bypass FCV Failure (FK-605B1 Closed)
MT-134	RHR to Letdown Valve Failure (HCV-142.1 Open)
MT-135	RHR Bypass Line Leak (Train A)
MT-136	RHR Sump Valves Fail to Open
MT-137	Containment Spray Pump Failure
MT-138	Containment Spray Pump Discharge Valve Failure
MT-151	Inadvertent Turbine Trip
MT-1519	Turbine First Stage Pressure Transmitter Failure (PT-446 Low)
MT-152	Turbine Protection Trip Failure
MT-154	Turbine Vibration
MT-155	Governor Valve Failure (GV-3 Closed)

MT-157	Turbine DEH Computer Failure
MT-17	Refueling Water Storage Tank Leak
MT-21	Component Cooling Water Pump Trip
MT-210	Seal Water Heat Exchanger Tube Leak
MT-22	Loss of CCW to RHR Heat Exchanger
MT-23	CCW Leak into the Service Water System
MT-24	Component Cooling Water Header Supply Valve Failure (Closed)
MT-25	Letdown Heat Exchanger Tube Leak
MT-26	Loss of CCW to RCP Thermal Barrier
MT-271	Letdown Temperature Controller Failure (TK-144 Low)
MT-272	Letdown Temperature Controller Failure (TK-144 High)
MT-28	Loss of CCW to the Reactor Coolant Pumps
MT-31	Circulating Water Pump Trip
MT-32	Main Condenser Tube Leak
MT-331	Hotwell Level Controller Failure (LC-1900 High)
MT-333	Hotwell Level Controller Failure (LC-1901 Low)
MT-34	Loss of Condenser Vacuum
MT-35	Loss of Condenser Vacuum Pump
MT-41	Power Cabinet Urgent Failure
MT-410	DRPI--Open or Shorted Coil
MT-411	Improper Bank Overlap
MT-412	Control Bank Rod Step Counter Failure
MT-413	Rod Speed Deadband Control Failure
MT-42	Logic Cabinet Urgent Failure
MT-431	Dropped Rod (One Rod)
MT-44	Stuck Rod
MT-45	Ejected Rod
MT-46	Uncontrolled Rod Motion
MT-47	Failure of Auto Rod Blocks to Block (C-11)
MT-48	TREF Failure
MT-49	DRPI--Loss of Voltage
MT-51	Letdown Isolation Valve Failure (1CS-11)
MT-5111	VCT Level Transmitter Failure (LT-112 High)
MT-5112	VCT Level Transmitter Failure (LT-115 Low)
MT-512	RCP Number 1 Seal Failure (RCP B)
MT-513	RCP Number 2 Seal Failure (RCP A)
MT-514	RCP Number 3 Seal Failure (RCP C)
MT-5151	Boric Acid Flow Transmitter Failure (FT-113 to 20 gpm)
MT-5152	Boric Acid Flow Transmitter Failure (FT-113 to 0 gpm)
MT-516	Boric Acid Filter Plugged
MT-5181	Seal Injection Flow Control Valve Failure (HC-186 Open)

MT-5182	Seal Injection Flow Control Valve Failure (HC-186 Closed)
MT-5201	Failure of Charging Flow Control Valve
MT-5202	Failure of Charging Flow Control Valve (Closed)
MT-523	High Temperature Divert Valve (TCV-143) Failure
MT-524	Charging Pump Suction From RWST Failure (115D Open)
MT-525	Charging Pump Mini Flow Valve Failure (1CS-182 Closed)
MT-526	Boric Acid Pump Trip
MT-527	Charging Line Containment Isolation Valve Failure
MT-5281	Charging Line Leak on Charging Pump Suction
MT-5282	Charging Pump Discharge Line Leak Before FT-122
MT-5283	Charging Line Leak Between FT-122 and 1CS-235
MT-5284	Charging Line Leak in Containment Before Regen HX
MT-5285	Charging Line Leak Between Regen HX and 1CS-492
MT-53	Letdown Line Leak Inside Containment
MT-54	Letdown Line Leak Outside Containment
MT-55	Charging Pump Trip
MT-56	Reactor Makeup Water Pump Trip
MT-571	Letdown Pressure Control Valve Failure (PK-145 Open)
MT-572	Letdown Pressure Control Valve Failure (PK-145 Closed)
MT-58	Loss of Normal Letdown
MT-59	VCT Divert Valve Control Failure (HUT)
MT-61	Station Blackout
MT-610	Loss of Unit Auxiliary Transformer
MT-612	Generator Output Breakers Fail to Trip
MT-615	Diesel Generator Governor Failure
MT-616	Diesel Generator Breaker Inadvertent Trip
MT-62	Loss of 120-VAC Uninterruptible Power (Power Supply SIII)
MT-63	Loss of 125-VDC Emergency Bus (DP 1B-SB)
MT-64	Loss of 6.9 KV Auxiliary Bus (1B)
MT-65	Loss of 6.9-KV Emergency Bus (1A-SA)
MT-66	Loss of a 125-VDC Nonvital Bus (DP 1A)
MT-67	Diesel Generator Failure
MT-68	Automatic Voltage Regulator Failure (High)
MT-69	Loss of Start-up Transformer
MT-710	Condensate Pump Trip (Pump A)
MT-711A	Motor Driven Auxiliary Feedwater Pump Trip
MT-712	Failure of Excess Condensate Dump Valve (Closed)
MT-712A	Turbine Driven Auxiliary Feedwater Pump Trip
MT-714	Condensate Storage Tank Leak
MT-715	Heater Drain Pump Trip (Pump B)
MT-719	Main Feedwater Pump Trip (Pump B)

MT-72	Condensate Booster Pump Trip (Pump B)
MT-720	Main Feedwater Pump Recirc Valve Failure (Pump 1B)
MT-721	Feedwater Flow Transmitter Failure (FT-466 Low)
MT-722	Feedwater Control Valve Position Failure (LCV-488 Open)
MT-723	Feedline Break Inside Containment
MT-724	Feedline Break Outside Containment
MT-725	Steam Generator Level Channel Failure (LT-496 Low)
MT-73	Turbine Driven Auxiliary Feedwater Pump Speed Control Oscillates
MT-76	Auxiliary Feedwater Flow Control Valve Failure (Open)
MT-771	Feedwater Bypass Valve Failure (Closed)
MT-772	Feedwater Bypass Valve Failure (Open)
MT-78	Turbine Driven Auxiliary Feedwater Flow Control Valve Failure (Closed)
MT-81	Steamline Break Inside Containment
MT-810	Steam Dump Control Failure (Closed)
MT-811	Mechanically Stuck Condenser Dump Valve (PCV-408 Open)
MT-812	Steam Dump Permissive (P-12) Failure
MT-814	Steam Failure to TDAFW Pump (1MS-72 Closed)
MT-815	Main Steam Header Break
MT-82	Steam Break Outside Containment
MT-83	Steam Header Pressure Detector Failure (PT-464 High)
MT-84	Steam-Line Flow Transmitter FT-494
MT-85	Steam Generator Pressure Transmitter Failure (PT-485 High)
MT-86	Steam Generator Relief Valve Failure (Open)
MT-87	MSIV Failure (S/G B Shut)
MT-88	Steam Generator Safety Valve Failure (Open)
MT-89	Atmospheric Steam Dump Valve Failure (PCV-408J Open)
MT-91	Source Range Instrument Failure (N31 High)
MT-911	Source Range Instrument Power Fuse Blown
MT-912	Intermediate Range Control Power Fuse Blown
MT-913	Power Range Control Power Fuse Blown
MT-92	Source Range Pulse Height Discriminator Failure
MT-93	Failure of Source Range High Voltage to Disconnect
MT-94	Source Range Channel High Voltage Failure
MT-95	Intermediate Range Channel Failure
MT-96	Intermediate Range Channel Gamma Compensation Failure
MT-97	Power Range Channel Detector Failure (Low)
MT-98	Power Range Channel Failure (Low)

Normal Operator Surveillance Tests (21)

NOST-001	Power Range Heat Balance
NOST-002	CVCS/SI System Operability
NOST-003	RHR Pump Operability
NOST-004	Containment Spray Operability
NOST-005	1A-SA Emergency Diesel Generator Operability
NOST-006	(Deleted)
NOST-007	Main Steam Isolation Valve and Main Feedwater Isolation Valve Operability Test
NOST-008	Daily Surveillance Requirements Modes 1 and 2
NOST-009	Daily Surveillance Requirements Modes 3 and 4
NOST-010	Reactor Coolant System Leakage Evaluation
NOST-011	Shutdown Margin Calculation
NOST-012	Calculation of Quadrant Power Tilt Ratio
NOST-013	Main Steam Isolation Valve Operability Test
NOST-014	Permissives P-6 and P-10 Verification
NOST-015	1B-SB Emergency Diesel Generator Operability
NOST-016	Turbine Mechanical Overspeed Trip Test
NOST-017	AFW Pump 1B-SB Operability Test - Quarterly
NOST-018	RHR Pump 1B-SB Operability
NOST-019	Reactor Coolant Pump Seals Controlled Leakage Evaluation
NOST-020	AFW Pump 1A-SA Operability Test - Quarterly
NOST-021	CCW System Operability - Quarterly
NOST-022	(Deleted)
NOST-023	Control Rod and Rod Position Exercise

Normal Operations Tests (6)

NOT-001	Plant Shutdown - GP-006
NOT-002	Plant Cooldown - GP-007
NOT-003	Plant Heatup - GP-002
NOT-004	Reactor Startup - GP-004
NOT-005	Plant Startup - GP-005
NOT-006	Recovery to Rated Power Following Reactor Trip

Real Time Test (1)

RTT-001	Computer Real Time Test

Steady State Tests (3)

SST-001	100 Percent Power Accuracy Test
SST-002	75 Percent Power Accuracy Test
SST-003	30 Percent Power Accuracy Test

Transient Tests (10)

TT-001	Manual Reactor Trip
TT-002	Simultaneous Trip of all Feedwater Pumps
TT-003	Simultaneous Closure of All Main Steam Isolation Valves
TT-004	Simultaneous Trip of All Reactor Coolant Pumps
TT-005	One Reactor Coolant Pump Trip
TT-006	Turbine Trip Below P-10
TT-007	Maximum Rate Power Ramp
TT-008	Maximum Size RCS Leak Inside Containment With Loss of Off-site Power
TT-009	Maximum Size Steam Leak Inside Containment
TT-010	Slow RCS Depressurization to Saturation Using PORV's and No SI

PERFORMANCE TEST ABSTRACT
FT-001

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Simulator Physical Fidelity Test

1.2 ANSI/ANS-3.5, 1985, Section 3.2

1.3 Regulatory Guide 1.149, Rev. 1, April 1987

2.0 AVAILABLE OPTIONS

N/A

3.0 TESTED OPTIONS

N/A

4.0 INITIAL CONDITIONS

N/A

5.0 TEST DESCRIPTION

This test is used to document the performance of Simulator Support Procedure SSP-216.2, Simulator Physical Fidelity. This test is done on an annual basis and compares the simulator and control room hardware.

6.0 BASELINE DATA/REFERENCES

Pictures of the plant control room and the simulator are reviewed annually and after any modification to simulated panels. Pictures of affected simulator and/or plant control room areas/panels will be retaken as changes warrant (at least annually).

7.0 **DATE PERFORMED/TEST RESULTS:** 01/20/91/Sat

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

8.1 SSR 90-0146, status board does not match control room.

Minimal impact as differences are minor.

8.2 SSR 90-0845, need chain across access.

No functional impact.

8.3 SSR 90-0917, main control sound generator.

Previously considered out of scope, the sound generator is being added in 1991.

8.4 SSR 90-1010, simulator demarcation.

Only minor differences noted, all being corrected.

8.5 SSR 90-1011, simulator meter legends and scales.

Only minor differences noted, all being corrected.

8.6 SSR 90-1012, simulator board labels.

Only minor differences noted, all being corrected.

8.7 SSR 90-1013, CSIP aux lube oil pump states light engraving.

Only minor differences noted, all being corrected.

8.8 SSR 91-0133, beeper on radio console.

Training valve assessment being performed for radio beeper.

8.9 SSR 91-0134, missing relay on relay panel.

No training impact, to be cosmetically simulated.

8.10 SSR 91-0135, miscolored SLBs on AEP-1.

Minor difference which is being corrected.

8.11 SSR 91-0136, RM-23 recorders.

Recorders capability being added in 1991, meters already work.

PERFORMANCE TEST ABSTRACT
MT-101

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSC-4, Inadvertent Containment Isolation Phase A (Train A)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an inadvertent containment Phase A isolation. Either Train A or B may be selected for the malfunction.

3.0 TESTED OPTIONS

Both trains are tested independently.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when both trains of containment isolation Phase A have been verified.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OMM-004, Post Trip/Safeguards Review

7.0 DATE PERFORMED/TEST RESULTS 6-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1013

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-22, Inadvertent Feedwater Isolation (Train A)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an inadvertent feedwater isolation. Feedwater isolation is actuated due to a spurious signal. Either Train A or B may be selected, or both feedwater trains may be selected, for isolation.

3.0 TESTED OPTIONS

Inadvertent feedwater isolation, Train A.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The malfunction causes both feedwater trains to receive a spurious isolation signal. This causes all feedwater isolation valves and bypass valves to close and trips all feedwater pumps. The spurious feedwater isolation signal also trips the turbine, resulting in a reactor trip. Auxiliary feedwater is actuated to supply feedwater to the SGs. The test is complete when all expected responses have occurred and all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134.01, Feedwater System

7.0 DATE PERFORMED/TEST RESULTS 6-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1014

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-12, Inadvertent Main Steam Line Isolation

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to an inadvertent main steam line isolation due to a spurious signal.

3.0 TESTED OPTIONS

Proper response to an inadvertent main steam line isolation.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

The malfunction causes all main steam isolation valves and their bypass valves to shut. Steam flow, header pressure and megawatt load decreases. RCS TAVE and pressure increases rapidly. SG PORVs and possibly SG safety valves open. The test is complete when proper main steam isolation is verified using OMM-004.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-126, Main Steam, Extraction Steam and Steam Dump Systems

6.3 OMM-4, Post Trip/Safeguards Review

7.0 DATE PERFORMED/TEST RESULTS 5-8-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1015

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 DSG-4, Diesel Generator Sequencer Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to the failure of the emergency sequencers to start loads. Either sequencer, or both, may be selected for a complete failure or only for a load block 9 failure.

3.0 TESTED OPTIONS

Failure of both sequencers to run.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the normally sequenced loads are verified not to start and all other indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-155, Diesel Generator Emergency Power System

6.3 AOP-025, Loss of One Emergency AC Bus or One Emergency DC Bus

7.0 DATE PERFORMED/TEST RESULTS 1-14-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 91-0022, ERFIS points EEE1726 and 1751 do not reflect actual breaker status

Minimal training impact. Points on safeguards display do not track actual. Actual status available in control room.

PERFORMANCE TEST ABSTRACT
MT-10161

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-7, Failure of Rod Blocks to Block (C-1)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of auto rod control rod blocks to stop rod motion. Any of the rod control system rod stop interlocks may be selected for the malfunction.

3.0 TESTED OPTIONS

Intermediate range (IR) high-flux (C-1) interlock failure.

4.0 INITIAL CONDITIONS

Mode 1 or 2, turbine ready to load.

5.0 TEST DESCRIPTION

The turbine is loaded and the TAVE-TREF mismatch error increases, with rods stepping out. The test is complete when the IR HIGH-FLUX annunciator alarms, rods continue stepping out, resulting in a reactor trip at the IR high- flux low reactor trip setpoint and all indications and alarms are verified correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

6.3 AOP-001, Malfunction of Reactor Control Systems

7.0 DATE PERFORMED/TEST RESULTS 10-15-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-10162

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-7, Failure of Rod Blocks to Block

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

Failure of auto rod control rod blocks to stop rod motion. Any or all of the Rod Control System rod blocks may be selected.

3.0 TESTED OPTIONS

Failure of rod blocks C-2 (overpower rod stop), C-3 (OT Delta-T rod stop) and C-4 (OP Delta-T rod stop).

4.0 INITIAL CONDITIONS

Mode 1, 100% power.

5.0 TEST DESCRIPTION

A power escalation is performed with rod control in automatic and the rod blocks defeated. Turbine runbacks from OT Delta T and OP Delta T are also defeated. The test is complete when rod motion is not interrupted as required by the associated rod block.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

6.3 Westinghouse Functionals 1364-867, 868, 872, 878

7.0 DATE PERFORMED/TEST RESULTS 10-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-10165

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-7, Failure of Rod Blocks to Block (C5)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a failure of auto rod control rod blocks to stop rod motion. Any of the rod control system rod block interlocks may be selected for this malfunction.

3.0 TESTED OPTIONS

Rod block C5, low turbine power load block auto rod withdrawal, interlock fails to block rod motion with rod control in AUTO.

4.0 INITIAL CONDITIONS

Mode 1 or 2, with the turbine ready to load.

5.0 TEST DESCRIPTION

If auto rod stop fails for C5, reactor power will continue to increase until operator action is taken. The test is complete when the TAVE-TREF error increases and rods are stepping out with C5 verified to be active.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

6.3 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 5-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1017

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Failure of permissive interlock P-14

1.2 ANSI/ANS-3.5, 1985, 3.1.2(22)

2.0 AVAILABLE OPTIONS

Failure of the OR gate in P-14 circuit will not allow the turbine to trip on Steam Generator Hi-Hi level, nor will the feedwater isolation or pump trip occur.

3.0 TESTED OPTIONS

Failure of P-14 circuit.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power.

5.0 TEST DURATION

The test is complete when all three steam generator levels are greater than the trip setpoint of 82.4 percent and all bistables, annunciators, and indications of the high level are verified. The turbine trip annunciator will come in for all three steam generators but no turbine trip will occur.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134.01, Feedwater system

7.0 DATE PERFORMED/TEST RESULTS 5-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-102

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Inadvertent Containment isolation, Phase B

1.2 ANSI/ANS-3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an inadvertent Phase B isolation.

3.0 TESTED OPTIONS

Spurious Containment isolation, Phase B, has occurred.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

When Containment Phase B isolation is actuated, component cooling water to the RCPs will be isolated and RCP motor temperatures will increase. The test is complete when OMM- 004, Attachment 9 for Phase B, is completed.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OMM-004, Post Trip/Safeguards Review

7.0 DATE PERFORMED/TEST RESULTS 5-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1031

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SIS-1, Safety Injection Failure (Train B, Inadvertent)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an inadvertent safety injection. Either Train A or B may be selected for an inadvertent or failure to initiate type of malfunction.

3.0 TESTED OPTIONS

An inadvertent safety injection occurs on Train B.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The reactor trips due to the safety injection signal and the safety injection first out annunciators alarm. The test is complete when OMM-004, Attachments 3-7 are completed and all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OMM-004, Post Trip/Safeguards Review

7.0 DATE PERFORMED/TEST RESULTS 6-7-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1032

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SIS-1, Safety Injection Failure (Train A, Fail to Initiate)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of Train A safety injection to actuate when needed. Either Train A or B may be selected.

3.0 TESTED OPTIONS

A failure of safety injection to initiate has occurred on Train A.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when OMM-004, Attachments 3-7 are completed, verifying proper SI alignment for Train B and a failure of Train A to initiate.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OMM-004, Post Trip/Safeguards Review

7.0 DATE PERFORMED/TEST RESULTS 6-7-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1041

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Reactor Trip Breakers Fail (BOTH, INADVERTENT, OPEN)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (19)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the inadvertent opening of the reactor trip breakers. Either trip breaker or both may be selected for the failure.

3.0 TESTED OPTIONS

An inadvertent opening of both reactor trip breakers occurs.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power.

5.0 TEST DESCRIPTION

The reactor trip breakers indicate open with all rods indicating fully inserted in the core. The test is complete when all alarms, indications and automatic functions associated with a reactor trip are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-103, Reactor Protection

7.0 DATE PERFORMED/TEST RESULTS 5-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1042

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Reactor Trip Breakers Fail (B Fails to Open in Auto)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (24)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of B RTB to trip on an automatic reactor trip signal. It also verifies that the breaker will trip on a manual trip signal. Either or both trip breakers may be selected with auto failure, manual failure or both modes may be selected for the failure to open subcategory.

3.0 TESTED OPTIONS

Reactor trip breaker B fails to open with an automatic reactor trip signal present.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power.

5.0 TEST DESCRIPTION

When a manual reactor trip is initiated, trip breaker A opens and trip breaker B remains closed. The test is complete when a manual reactor trip is actuated and trip breaker B opens.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-103, Reactor Protection System

7.0 **DATE PERFORMED/TEST RESULTS** 5-4-90/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
MT-106

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CNS-1, False Containment Spray Actuation

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a spurious containment spray actuation.

3.0 TESTED OPTIONS

A spurious containment spray actuation is received.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The malfunction causes both containment spray pumps to start. Discharge pressures increase but flow remains at zero, since the pump discharge valves are closed. The test is complete when all indications and alarms are verified to be correct, and the malfunction is cleared with the containment spray actuation signal reset.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-112, Containment Spray System

6.3 OST-1119, Containment Spray Operability, Train B

6.4 OST-1129, Containment Spray ISI Valve Test

7.0 DATE PERFORMED/TEST RESULTS 5-8-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1071

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-22, Turbine Runback Failure (Erroneous Runback)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

An erroneous turbine runback or a failure to runback can be selected.

3.0 TESTED OPTIONS

Erroneous turbine runback.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

An erroneous turbine runback occurs. All the indications of a runback are confirmed. After the plant response to a runback is verified, the runback is determined to be erroneous by the absence of any valid turbine runback signal. The test is complete when an attempt is made to increase load and the runback continues.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-131.01, Main Turbine

7.0 DATE PERFORMED/TEST RESULTS 5-8-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1072

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-22, Turbine Runback Failure (Failure to Runback)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The procedure tests proper response to a failure of the turbine to runback when required.

3.0 TESTED OPTIONS

Turbine runback failure.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The main feedwater pump trips, the turbine does not runback and a reactor trip occurs on low-low SG level. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-131.01, Main Turbine

7.0 DATE PERFORMED/TEST RESULTS 1-24-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-111

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 AIR-1, Loss of Instrument Air to the Turbine Building

1.2 ANSI/ANS 3.5, 1985 3.1.2 (2)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a loss of instrument air. The containment, auxiliary building or turbine building may be selected for the failure. A loss of service air may also be selected.

3.0 TESTED OPTIONS

Turbine building loss of instrument air.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

The test is complete when all components in the turbine building, on the affected air header, fail to the required positions and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-017, Loss of Instrument Air

6.3 SFD-2165-S-0542, 0543, 544, 0545, 0546, 0548, 0551

7.0 DATE PERFORMED/TEST RESULTS 1-15-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1110

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-10, Pressurizer Level Control Band Shift Down

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer level control band shift down from normal reference. On a plant shutdown from 100 percent, this will cause the pressurizer setpoint to stay at 60 percent, until TAVE is approximately 578°F and then ramp down with TAVG.

3.0 TESTED OPTIONS

Pressurizer is filled to 60 percent level and pressurizer level control band shifts to 547°F.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Normal shutdown of the plant is in progress. The test is complete when TAVG reaches approximately 577°F and the level setpoint on the trend recorder starts to come down with TAVG.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-111A

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-1, Pressurizer Steam Space Leak

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (c)

2.0 AVAILABLE OPTIONS

A selectable leak rate from 0 to 800 gpm and a ramp time of 0 to 3600 seconds.

3.0 TESTED OPTIONS

A leak of 800 gpm is entered with no ramp time.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is a leak on a pressurizer safety valve flange. Pressure will rapidly drop in the RCS and safety injection will actuate on pressurizer low pressure. The pressurizer will fill up as a void forms in the reactor vessel head. The test is complete when the containment response is verified and the expected annunciators are received and verified.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.1 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 10-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-112

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 AIR-1, Loss of Instrument Air to the Reactor Auxiliary Building

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (2)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a loss of instrument air. The Containment, Auxiliary Building, or Turbine Building may be selected for the failure. In addition, service air may also be selected.

3.0 TESTED OPTIONS

Loss of instrument air to the Auxiliary Building.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 90% power, with feedback loops in service.

5.0 TEST DESCRIPTION

The test will verify the automatic actions that occur in the instrument and service air systems as pressure decreases. The leak is isolated to the Auxiliary Building using local operator actions. The test is complete when all indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-017, Loss of Instrument Air

6.3 SFD 2165-S-0800 and 0801

7.0 DATE PERFORMED/TEST RESULTS 9-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-112A

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-2, Pressurizer Spray Valve Failure

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response for a failure of either or both pressurizer spray valves, to any position from 0 to 100 percent open, with manual control not available.

3.0 TESTED OPTIONS

One spray valve failed 100 percent open.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 75% power.

5.0 TEST DESCRIPTION

The spray valve fails open and pressurizer pressure decreases to the reactor trip setpoint, where the reactor coolant pump, in the loop associated with the failed valve, is secured. After the pump is secured, pressure stabilizes and the test is complete when all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 10-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-113

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 AIR-1, Loss of Instrument Air to Containment Building

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (2)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a loss of instrument air. The Containment, Auxiliary Building or Turbine Building may be selected for the failure. In addition, a loss of service air may also be selected.

3.0 TESTED OPTIONS

Loss of instrument air to Containment.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Listed components fail to the proper position, instrument air is isolated to Containment and the location of the leak is verified. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-151.01, Compressed Air

6.3 AOP-017, Loss of Instrument Air

7.0 DATE PERFORMED/TEST RESULTS 1-5-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1131

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-3, Pressurizer Relief Valve Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

Any of the three pressurizer power relief valves can be failed from 0 to 100 percent open. Interlock P-11 can be selected as either operable or inoperable during the failure. Interlock P-11 sends a shut signal to the power relief valves when pressure is less than 2000 psig. If P-11 is operable, the valve will shut at 2000 pounds RCS pressure or the control switch will function to shut the valve. If P-11 is inoperable, the relief valve cannot be shut.

3.0 TESTED OPTIONS

Failure of one PORV fully open, with P-11 interlock operable.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 75% power.

5.0 TEST DESCRIPTION

The malfunction will cause pressure to rapidly decrease to 2000 psig where the P-11 interlock will shut the PORV. If pressure decreases to 1960 psig before the PORV is fully closed, the reactor will trip. After the PORV closes, pressure will stabilize and start increasing.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 6-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1132

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-3, Pressurizer Relief Valve Failure (Without P-11 Interlock)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

Any of the three pressurizer power relief valves can be failed from 0 to 100 percent open. Interlock P-11 can be selected as either operable or inoperable during the failure. Interlock P-11 sends a shut signal to the power relief valves when pressure is less than 2000 psig. If P-11 is operable, the valve will shut at 2000 pounds RCS pressure or the control switch will function to shut the valve. If P-11 is inoperable, the relief valve cannot be shut.

3.0 TESTED OPTIONS

Failure of one PORV fully open, with P-11 interlock inoperable.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 75% power.

5.0 TEST DESCRIPTION

The malfunction will cause pressure to rapidly decrease to 2000 psig where the PORV is verified to remain open with pressure less than the P-11 setpoint. Pressure continues to decrease until the reactor trips and safety injection occurs. The PORV block valve is shut to isolate the open PORV, allowing pressure to start increasing.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-114

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-4, Pressurizer Safety Valve Failure (Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (d)

2.0 AVAILABLE OPTIONS

The malfunction allows failing each of the pressurizer safety valves from 0 to 100 percent open.

3.0 TESTED OPTIONS

One pressurizer safety valve is failed fully open.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Pressurizer pressure rapidly decreases. Automatic actions expected to occur as a result of the decreasing pressure are verified. As pressure decreases, a bubble forms in the vessel head, as indicated on reactor vessel level indication. Pressurizer level increases until solid. System pressure stabilizes as SI flow equals break flow and incore temperatures trend downward.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 10-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1151

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-5, Pressure Channel Failure (PT-444 High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer pressure control channel failure. Either control channel may be selected to fail to a selected value of 1700 to 2500 psia.

3.0 TESTED OPTIONS

Pressurizer control channel PT-444 fails high.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The malfunction causes the spray valves and PORV to open with a drop in RCS pressure. When P-11 is reached, the PORV closes, but pressure continues to decrease due to the open spray valve. The test is complete when the master pressure controller is selected to MANUAL to restore pressure to the controller setpoint, and all alarms and indications are verified correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1152

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-5, Pressure Channel Failure (PT-445 Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer pressure control channel failure. Either control channel may be selected to fail to a selected failed value of 1700 to 2500 psia.

3.0 TESTED OPTIONS

Pressurizer protection channel PT-445 fails low.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all indications and alarms related to the malfunction are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1161

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-6, Pressurizer Pressure Channel Failure (PT-456, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a pressurizer pressure protection channel failure. Any of the protection channels may be selected to fail in a range of 1700 to 2500 psia.

3.0 TESTED OPTIONS

Pressurizer protection channel PT-456 failure, high.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the various bistables and alarms associated with the malfunction are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1162

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-6, Pressurizer Pressure Channel Failure (PT-457, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer pressure protection channel failure. Any of the protection channels may be selected to fail in a range of 1700 to 2500 psia.

3.0 TESTED OPTIONS

Pressurizer protection channel PT-457 failure, low.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the various bistables and alarms associated with the malfunction are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1171

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-7, Pressurizer Level Channel Failure (LT-459 Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer protection level channel failure. Any of the level channels may be selected to fail to a designated failed value of 0 to 100 percent.

3.0 TESTED OPTIONS

Pressurizer level channel LT-459 fails low.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The malfunction, with the assumption that LT-459 is selected as the controlling channel, causes the associated letdown isolation valve to close, securing normal letdown. Pressurizer heaters trip off and charging flow and pressurizer level increase. The test is complete when LT-459 is removed as the selected controlling channel, and charging and letdown respond to trend pressurizer level back to normal.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1172

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-7, Pressurizer Level Channel Failure (LT-459 High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer protection level channel failure. Any of the level channels may be selected to fail to a designated failed value of 0 to 100 percent.

3.0 TESTED OPTIONS

Pressurizer level channel LT-459 fails high.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The malfunction, with the assumption that LT-459 is selected as the controlling channel, causes the backup heaters to energize and charging flow will decrease along with pressurizer level. The test is complete when LT-459 is removed as the selected controlling channel, and charging and letdown respond to trend pressurizer level back to normal.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-118

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 PRS-8, Pressurizer Backup Heater Groups A and B Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a pressurizer backup heater group failure. Either or both backup groups may be selected for the failure.

3.0 TESTED OPTIONS

Failure of backup heater groups A and B.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The backup heaters are verified to be off by a sharp decrease in pressure on the ERFIS plot. The malfunction is then removed and the heaters are verified to be energized by the ERFIS plot and amp indication. The test is complete when all indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 1-15-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-12

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SWS-1, NSW Pump Trip and Loss of NSW

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (6)

2.0 AVAILABLE OPTIONS

Either, or both NSW pumps may be tripped

3.0 TESTED OPTIONS

SWS-1A and SWS-1B.

4.0 INITIAL CONDITIONS

Mode 1, with NSW Pump 1A in service.

5.0 TEST DESCRIPTION

The test will verify the correct automatic actions that occur when a normal service water pump trips. The test goes on to trip the remaining NSW pump and verifies that the temperatures increase on components cooled by the NSW system. Emergency service water will auto start and supply the ESW loads on the loss of NSW.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-139, Service Water System

6.3 AOP-022, Loss of Service Water

7.0 DATE PERFORMED/TEST RESULTS 4-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1210

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-10, RCP A, B, C, High Vibration

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to Reactor Coolant Pump (RCP) high vibration. Vibration levels may be selected in a range of 0 to 30 mils, on up to three pumps simultaneously. The start time for the malfunctions can be delayed to allow monitoring of each RCP separately.

3.0 TESTED OPTIONS

Each RCP with high vibration, developed at three minute intervals.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The shaft vibrations for the selected RCP increases. The frame vibration level increases about one-fifth as much as the shaft vibration. Possible RCP vibration high alarm at 15 mils on shaft and 3 mils on frame vibration. The test is complete when the indicators and alarms show shaft and frame vibration increasing and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 **DATE PERFORMED/TEST RESULTS** 10-12-90/SAT

8.0 DEFICIENCES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1211

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-1, LOCA Within Capacity of the SI Pumps

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (C)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a loss of reactor coolant within the capacity of the charging pumps. More than one LOCA may be active from any of the loop's hot or cold legs, however, the leak rate is based on normal operating pressure and will vary as pressure varies. The leak rate is selectable with a break ranging from 0 to 4.3 sq. ft.

3.0 TESTED OPTIONS

Loop 1 cold leg develops a break of 6E-4 sq. ft. (approximately 82 gpm) at normal operating pressure.

4.0 INITIAL CONDITIONS

Mode 1, and approximately 100% power.

5.0 TEST DESCRIPTION

Pressurizer pressure and level decreases and charging increases to maximum capability. The test is complete when pressure and level in the pressurizer stabilizes as charging flow matches break flow.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-14-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1211A

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RTC-1, RCS Fuel Rod Breach

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (14)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to an RCS fuel breach. The indications of a fuel breach come from PEP-101, Attachments 1 and 2. The percent of fuel failure may be selected with a range of 0-100 percent.

3.0 TESTED OPTIONS

Fuel failure of 2 percent.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when activity levels in the RCS increase, letdown liquid radiation levels increase and high radiation alarms are received.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-032, High RCS Activity

6.3 PEP-101, Emergency Classification and Initial Emergency Actions

6.4 NUREG-0654, Appendix 1

7.0 **DATE PERFORMED/TEST RESULTS** 1-23-91/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

SSR 90-0120, Radiation monitor 3502A-SA locked up

Minimal training impact due to RM-11 indication still available.

PERFORMANCE TEST ABSTRACT
MT-1212

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CS-1, LOCA Within Capacity of the SI Pumps

1.2 ANSI/ANS 3.5, 1985, 3.1.2

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a large break loss of coolant accident within the capacity of the safety injection pumps. More than one LOCA may be active from any combination of cold or hot legs, however, the leak rate will not change due to the effect of the additional LOCA. The leak size may be selected with a range of 0 to 4.3 sq. ft., based on normal pressure, and the leak rate will vary as pressure varies.

3.0 TESTED OPTIONS

Loop 2 cold leg break of 5E-2 sq. ft.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when RCS pressure stabilizes where SI flow equals break flow, a slow RCS cooldown starts due to the SI flow, and all indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-14-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1212A

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SIS-2, RCS Leakage into an Accumulator

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to RCS leakage into an SI accumulator. Any combination of up to three accumulators may be selected with a leak rate of 0 to 50 gpm.

3.0 TESTED OPTIONS

SI Accumulator A, Loop 1, check valve has 50 gpm RCS inleakage

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power, with rods in automatic.

5.0 TEST DESCRIPTION

The test is complete when SI Accumulator A has a level and pressure increase and all associated alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-110, Safety Injection System

7.0 DATE PERFORMED/TEST RESULTS 5-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1213

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-10, RCS Vessel Flange Leak

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a Reactor Coolant System vessel flange leak. Leakage from the inner seal ring may be varied from 0 to 30 gpm.

3.0 TESTED OPTIONS

Reactor vessel flange leakage from the inner seal at 30 gpm.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The malfunction causes increased temperature indication and an alarm for reactor vessel flange leakoff. The test is complete when the vessel leak is isolated. A stabilized pressurizer level and a decreasing flange leakoff line temperature are used to verify that the leak has stopped.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 1-24-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1214

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-11, RCP Bearing Oil Reservoir Leak

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an RCP oil leak on its bearing oil reservoir. Any combination of up to three RCPs may be selected for the leak.

3.0 TESTED OPTIONS

Reactor Coolant Pump A bearing oil reservoir oil leak.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

When the oil leak occurs on the RCP, increased bearing oil temperatures and shaft vibrations are observed. The test is complete when all indications and alarms are verified correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 10-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1215

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-12, RCP Thermal Barrier Leak into CCW System

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a reactor coolant pump thermal barrier heat exchanger leak. Any of the three RCPs may be selected for the leak with a leak rate of 0 to 120 gpm.

3.0 TESTED OPTIONS

RCP B thermal barrier heat exchanger leak, 120 gpm.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

CCW surge tank level increases with decreasing seal water flow to the unaffected RCP's seals. Annunciator alarms are received and the thermal barrier return flow control valve closes. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-15-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1216

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-13, RCS flow transmitter failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

Failure of up to six of nine RCS flow transmitters over the range of 0 to 120 percent.

3.0 TESTED OPTIONS

Failure of one RCS loop flow transmitter low with the plant above the P-8 setpoint.

4.0 INITIAL CONDITIONS

Mode 1, 100% power

5.0 TEST DESCRIPTION

When the flow transmitter fails low, the 1/3 low flow logic causes an RCS loop low flow ALERT annunciator alarm, without causing a Reactor trip.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS: 10-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1221

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SGN-5, Steam Generator Tube Leak (B SG)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (a)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a tube leak in the steam generators. Any SG may be selected for the malfunction with a leak rate of 0 to 600 gpm. The leak rate will vary as differential pressure across the tube varies.

3.0 TESTED OPTIONS

Steam Generator B develops a 90 gpm tube leak.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Steam generator B shows a slight water level increase along with secondary radiation monitors revealing increasing radiation levels. The test is complete when the plant stabilizes with a slightly lower feedwater flow to B SG, SG water level returns to program level and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-15-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1222

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SGN-5, Steam Generator Tube Rupture (A SG)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (a)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a design basis tube rupture in the steam generators. Any SG may be selected for the malfunction with a leak rate of 0 to 600 gpm, varying as differential pressure across the SG tube varies.

3.0 TESTED OPTIONS

Steam Generator A with a 600 gpm tube rupture.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Pressurizer level and RCS pressure decreases, and secondary activity increases with subsequent high secondary radiation alarms. SG water level increases in A SG with a subsequent decrease in feedwater flow. A reactor trip on OT Delta-T or low pressure will occur, followed by safety injection actuation. The test is complete when all alarms and indications of a design basis tube rupture are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-15-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1231

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-2, RCP Trip from 100 Percent Power (C RCP)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (4)

2.0 AVAILABLE OPTIONS

The malfunction allows the tripping of individual reactor coolant pumps or tripping of all of the reactor coolant pumps simultaneously. The cause of the trip can be selected as either underfrequency, undervoltage or overcurrent.

3.0 TESTED OPTIONS

Overcurrent trip of C RCP.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is compared to actual plant data taken from the trip of a reactor coolant pump from 100 percent power. Loop flow goes to zero and increases to some low value as reverse flow begins. Pump parameters are observed to be indicating the pump tripped. The proper bistables and annunciators are verified to be lit for two loop operation. Computer points are captured during this test and compared to actual plant data for accuracy.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 SHNPP "C" RCP trip event of 6-17-87

7.0 DATE PERFORMED/TEST RESULTS 5-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1232

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-2, Reactor Coolant Pump Trip (C RCP)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (4)

2.0 AVAILABLE OPTIONS

The malfunction allows tripping any of the 3 reactor coolant pumps or all at once. The fault can be selected as underfrequency, undervoltage or overcurrent.

3.0 TESTED OPTIONS

Trip of C RCP due to overcurrent.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 30% power.

5.0 TEST DESCRIPTION

The reactor coolant pump trip occurs at less than 50 percent power to prevent a reactor trip from occurring. All annunciators that are required to alarm are confirmed. Loop C hot leg temperatures goes to cold leg temperature as loop flow decreases to zero and reverse flow begins. Loops A and B differential temperatures increase as they pick up the load lost from Loop C and the loops average coolant temperature remains unchanged. The plant stabilizes at approximately the same power level as before the transient.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-16-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-124

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-3, Reactor Coolant Pump Trip (Locked Rotor)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (4)

2.0 AVAILABLE OPTIONS

Locked rotor on Reactor Coolant Pump A, B or C.

3.0 TESTED OPTIONS

The locked rotor is on B Reactor Coolant Pump.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The reactor quickly trips due to low flow in B Loop. Loop average temperature and pressurizer pressure initially increase until the trip. Several annunciators are received and the remainder of the test verifies the annunciators are correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-1--90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-125

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-4, Reactor Coolant Pump Shaft Break Accident

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (4)

2.0 AVAILABLE OPTIONS

Shaft break on Reactor Coolant Pump A, B or C.

3.0 TESTED OPTIONS

The shaft break is on B Reactor Coolant Pump.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

When the shaft breaks, the reactor coolant pump remains running as flow in the loop decreases. This causes a pressure and temperature increase in the RCS, until the plant trips on reactor coolant loop low flow. All bistables and annunciators associated with the transient are verified to occur.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-1--90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-126

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-5, Variable RCS boron concentration

1.2 ANSI 3.5 Reference N/A

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a change in RCS boron concentration in a range of 0-3000 ppm with a selectable rate of 0-3600 seconds.

3.0 TESTED OPTIONS

Plant/system response to an inadvertent RCS dilution of 14 ppm in 18 seconds.

4.0 INITIAL CONDITIONS

Mode 1, 100% power, BOL, Rod Control System in automatic.

5.0 TEST DESCRIPTION

Following the activation of the dilution accident, which causes the RCS boron concentration to decrease by 14 ppm in 18 seconds, an increase in Tavg is observed. When a Tavg-Tref mismatch, sufficient to cause control rod movement is sensed, the rod control system responds by inserting control rods to restore Tavg to the Tref setpoint. When this occurs, the test is terminated.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS: 10-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1271

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-6, RCS Control RTD Failure (TE-411B, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a pressurizer RCS control RTD on the hot leg failing. The control RTD affects the Pressurizer Level Control System, Steam Dumps, Rod Insertion Limit Computer, Rod Control and the Turbine Loading Control Interlock C16. Any combination of up to six RTDs may be selected for the failure with a range of 530 to 630°F.

3.0 TESTED OPTIONS

Failure of narrow-range hot leg RTD TE-411B, high, at 630°F.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 80% power, with rods less than 186 steps (100 percent power Rod Insertion Limit).

5.0 TEST DESCRIPTION

The high failure of the RTD causes rods to step in and pressurizer level control to increase charging, due to TAVE indicating high. The test is complete when rods are placed in manual to restore TAVE to TREF and the failed loop is bypassed with the channel defeat switch located on the MCB.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 5-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1281

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-7, RCS Protection RTD Failure (TE-421B, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an RCS protection grade hot leg RTD failure. The RTD affects the OT Delta-T and OP Delta-T trips, Rod Stop/Runback, Low TAVE Feedwater Isolation and Low-Low TAVE Interlock P-12. Any combination of six RTDs may be selected for the failure, with a range of 530 to 630°F.

3.0 TESTED OPTIONS

Narrow-range RTD TE-412B failure, low, at 530°F.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 1-5-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1282

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-7, RCS Protection RTD Failure (TE-422B, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an RCS protection grade hot leg RTD failure. The failure affects the OT Delta-T and OP Delta-T trips, Rod Stop/Runback, Low TAVE Feedwater Isolation and Low-Low TAVE Interlock P-12. Any combination of six RTDs may be selected for the failure, with a range of 530 to 630°F.

3.0 TESTED OPTIONS

Narrow-range RTD TE-422B failure, high, at 630°F.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 10-15-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-129

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-8, RCS WR Pressure Transmitter Failure (PT-403, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The procedure tests proper response to a failure high of the RCS wide range pressure transmitter while on RHR.

3.0 TESTED OPTIONS

RCS wide range pressure transmitter (PT-403) failure, high.

4.0 INITIAL CONDITIONS

Mode 4, on RHR.

5.0 TEST DESCRIPTION

RCS pressure increases to 800 psig. Two RHR loop suction valves close and two remain open. RHR pump discharge pressure, flow and amperage decreases. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 DATE PERFORMED/TEST RESULTS 5-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-13

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 HVA-1, Containment Fan Cooler Trip

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the response to a containment fan cooler trip due to a 42 relay failure. Any of the containment fan coolers "A" fan may be selected for the malfunction, as well as any combination of the four fans to fail simultaneously.

3.0 TESTED OPTIONS

Containment Fan Cooler AH-2 "A" Fan failure.

4.0 INITIAL CONDITIONS

Mode 1 with Train A fan coolers in service.

5.0 TEST DESCRIPTION

The test is complete when the indications of the fan trip are received and all indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 Control Wiring Diagrams 2166-B-401, 2673, 2674, 2675, 3062, 3067, 3068, 3083

7.0 DATE PERFORMED/TEST RESULTS 1-22-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0982, ERFIS points for containment fan coolers

Minimal training impact as these points are not normally monitored.

PERFORMANCE TEST ABSTRACT
MT-131

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-1, RHR Pump Trip (Pump A)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to the trip of an RHR pump, while on RHR, due to an overcurrent condition. Either train's pump may be selected for the failure.

3.0 TESTED OPTIONS

RHR Pump A trip while in cooldown lineup.

4.0 INITIAL CONDITIONS

Mode 4, on RHR.

5.0 TEST DESCRIPTION

The affected train's flow decreases to zero. RCS cooldown rate and letdown flow decrease. If the plant is solid, an increase in pressurizer pressure could occur. The test is complete when the pump trips and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 **DATE PERFORMED/TEST RESULTS** 1-20-91/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
MT-1321

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-2, RHR HX Control Valve Failure (FCV-603A, Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the inadvertent closure of an RHR HX Outlet FCV. Either train's FCV may be selected for the failure, with a failed position of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

RHR HX A Outlet FCV (FCV-603A) failure, fully closed.

4.0 INITIAL CONDITIONS

Mode 4, on RHR, RHR HX A Bypass Valve Controller FK-605A in automatic.

5.0 TEST DESCRIPTION

Flow through the RHR heat exchanger decreases. The bypass valve opens to maintain proper RHR pump flow and the RCS cooldown rate decreases or stops. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 DATE PERFORMED/TEST RESULTS 1-20-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1322

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-2, RHR HX Control Valve Failure (FCV-603B, Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to an RHR HX outlet FCV failing open while on RHR. Either heat exchanger outlet valve may be selected to a failed position of 0 to 100 percent, closed to open. Manual control of the valve remains available.

3.0 TESTED OPTIONS

FCV-603B, RHR HX "B" outlet FCV, fails fully open.

4.0 INITIAL CONDITIONS

Mode 4, on RHR and RHR HX B bypass valve FK-603B is in automatic.

5.0 TEST DESCRIPTION

Flow through the RHR heat exchanger increases. The cooldown rate increases due to the higher flow. CCW temperature increases, possibly actuating various temperature alarms. The test is complete when all indications and alarms are verified correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 DATE PERFORMED/TEST RESULTS 1-18-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1331

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-3, RHR Hx Bypass FCV Failure (FK-605A1, Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a RHR heat exchanger FCV failing while on RHR. Manual control is available for the valve. Either train of RHR may be selected for the failure, with a failed position of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

FK-605A (1RH-20) failure, fully open.

4.0 INITIAL CONDITIONS

Mode 4, on RHR with FK-605A in automatic.

5.0 TEST DESCRIPTION

Flow through the RHR heat exchanger decreases. Total flow through the RHR train increases to maximum allowed by system pressure. RCS cooldown rate decreases or stops. Letdown from RHR decreases, causing an increase in pressurizer pressure. The test is complete when the bypass FCV is selected to manual and restored to its initial position, RHR parameters return to their initial values and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 DATE PERFORMED/TEST RESULTS 1-19-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1332

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-3, RHR HX Bypass FCV Failure (FK-605B1, Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the RHR HX bypass FCV failing while in Mode 4, on RHR. Manual control is available to operate the valve. Either RHR train may be selected with a failed position range of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

FK-605B (1RH-58) failed to the fully shut position.

4.0 INITIAL CONDITIONS

Mode 4, on RHR, and RHR bypass valve FK-605B in automatic.

5.0 TEST DESCRIPTION

The malfunction causes flow through the RHR heat exchanger to increase. The increase in flow causes the cooldown rate to increase. Total flow through the RHR system decreases. CCW system temperature may increase. The test is complete when the bypass FCV is placed in manual and re-opened to it's initial position, and all RHR pump parameters return to their initial values.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 DATE PERFORMED/TEST RESULTS 1-18-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-134

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-4, RHR to Letdown Valve Failure (HCV-142.1, Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response for the RHR to letdown valve failure while in Mode 4 and on RHR. HCV-142 fails to selected position, from 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

HCV-142 (1CS-28) fails fully open while on RHR.

4.0 INITIAL CONDITIONS

Mode 4, on RHR, with PK-145.1 (1CS-38), letdown pressure, in automatic.

5.0 TEST DESCRIPTION

The malfunction causes letdown flow through the CVCS to increase with the RHR system in service. The test is complete when PK-145.1 opens to reduce letdown pressure and flow, and RCS pressure stabilizes.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

6.3 AOP-020, Loss of Residual Heat Removal

7.0 DATE PERFORMED/TEST RESULTS 1-20-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-135

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-5, RHR Bypass Line Leak (Train A)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7)

2.0 AVAILABLE OPTIONS

The malfunction verifies the proper response to a leak on either Train A or B RHR heat exchanger bypass line, outside of containment, with a range of 0 to 3500 gpm.

3.0 TESTED OPTIONS

A 3500 gpm leak on RHR Train A heat exchanger bypass line.

4.0 INITIAL CONDITIONS

Mode 4, on RHR.

5.0 TEST DESCRIPTION

When the leak occurs, indicated RHR system pressure and flow decrease and pump amperage increases, due to the increase in total pump flow. Pressurizer pressure and level decrease as inventory is lost. The test is complete when associated area radiation monitors are verified to alarm and the leak is verified to be isolated in accordance with the Abnormal Operating Procedure.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 1-20-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-136

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RHR-6, RHR Sump Valves Fail to Open

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7) and (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the RHR sump valves failing to open on RWST low-low level, with a safety injection present.

3.0 TESTED OPTIONS

RHR sump valves fail to open on a low-low RWST level, with a safety injection signal present.

4.0 INITIAL CONDITIONS

Mode 1, followed by a reactor trip and safety injection.

5.0 TEST DESCRIPTION

The RHR pumps continue taking a suction on the RWST after the RWST low-low level setpoint has been reached. The test is complete when it is verified that RWST level continues to decrease, and all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-111, Residual Heat Removal System

7.0 DATE PERFORMED/TEST RESULTS 1-20-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-137

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CNS-2, Containment Spray Pump Failure

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a failure of a containment spray pump. Either pump may be selected for the malfunction.

3.0 TESTED OPTIONS

Two runs are performed. One with one running containment spray pump, and one with no containment spray pumps.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

A design basis LOCA is initiated. When containment pressure reaches its peak a hard copy of the trend is begun. Each run is complete when the A (and B) containment spray pump does not start and all data collection, indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-112, Containment Spray System

6.3 OST-1119, Containment Spray Operability, Train B

6.4 OST-1129, Containment Spray System ISI Valve Test

7.0 DATE PERFORMED/TEST RESULTS 10-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-138

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CNS-3, Containment Spray Pump Discharge Valve Failure

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a containment spray pump discharge valve failure. Either or both pump discharge valves may be selected for the failure.

3.0 TESTED OPTIONS

Containment Spray Pumps A and B discharge valves fail to open with a containment spray actuation present.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

On receipt of a containment spray actuation signal, the discharge valves fail to open. Manual attempts to open the discharge valves are made from the control board. The test is complete when the malfunction is cleared and the discharge valves open.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-112, Containment Spray System

6.3 OST-1119, Containment Spray Operability, Train B

6.4 OST-1129, Containment Spray ISI Valve Test

7.0 DATE PERFORMED/TEST RESULTS 10-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-151

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-1, Inadvertent Turbine Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (15)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to an inadvertent turbine trip due to a faulty mechanical trip valve.

3.0 TESTED OPTIONS

Inadvertent turbine trip.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The malfunction causes the reactor to trip and steam dumps to open in an attempt to maintain no-load TAVE. Steam generator parameters respond to the transient and the AFW pumps operate as required. The test is complete when all alarms, bistable lights and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 1-22-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-1519

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-17, Turbine First-State Pressure Transmitter Failure (PT-446, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a first-stage pressure transmitter failure. Either of two transmitters may be selected for the failure in a range of 0 to 860 psig.

3.0 TESTED OPTIONS

Turbine first-state pressure transmitter PT-446 fails low.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 75% power, the feedback loops in service and the TURBINE FIRST-STATE PRESSURE CONTROL SELECTOR selected to PT-446.

5.0 TEST DESCRIPTION

With the failed channel selected for control and the TAVE-TREF mismatch reading high, rods step in reducing reactor power and TAVE. The governor valves open in an attempt to maintain load. The test is complete when PT-447 is selected for control, rods step out to restore TAVE and return the plant to the initial power level, and all parameters return to normal.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-131.05, Digital Electrohydraulic Fluid System

7.0 DATE PERFORMED/TEST RESULTS 6-15-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-152

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-2, Turbine Protection Trip Failure

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the turbine to trip when a reactor trip occurs above P-7. At a high power level, the plant will quickly reach the rate compensated low steam line pressure SI setpoint, and a steam line isolation will occur. For this reason, the test is run at a lower power level to allow the DEH Control System time to react and attempt to maintain megawatt output before C-16 and a steam line isolation occurs.

3.0 TESTED OPTIONS

The turbine automatic trip circuit has failed, however, the manual turbine trip is still operable.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 16% power.

5.0 TEST DESCRIPTION

Following the initiation of the malfunction, the reactor is manually tripped. The turbine will fail to trip when required by protection logic. If the turbine is not tripped manually, the problem which induced the trip will be compounded by the turbine continuing to draw steam. The test is complete when all alarms, bistable lights and indications of the turbine failure to trip are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134.04, Main Turbine

7.0 DATE PERFORMED/TEST RESULTS 6-8-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-154

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-4, Turbine Vibration

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to turbine bearing vibration. Turbine bearings associated with the HP or either LP turbine may be selected, with a vibration amplitude in the range of 0 to 20 mils.

3.0 TESTED OPTIONS

LP turbine, number 3 bearing, with vibration of 12 mils.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The failed bearing vibrates at set amplitude and the bearings on each side of the selected bearing also vibrate, but at a decreasing amplitude as distance from the failed bearing increases. The test is complete when all alarms and indications for the failure are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-131.01, Main Turbine

6.3 AOP-006, Turbine Generator Trouble

7.0 DATE PERFORMED/TEST RESULTS 10-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-155

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-5, Governor Valve Failure (GV-3, Closed)

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a governor valve failing closed while operating at 100 percent power. Any of the governor or throttle valves may be selected for the failure, with a range of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

GV-3 failure, fully closed.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

A decrease in first-stage pressure occurs, with decreasing turbine load and additive governor valve position. Steam dumps actuate as required to reduce steam pressure and the control rods insert to reduce the subsequent TAVE to TREF mismatch. The test is complete when the DEH System compensates by repositioning the other governor valves to a new position, and all indications, alarms and bistables are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-131.05, Digital Electrohydraulic Fluid System

6.3 AOP-015, Secondary Load Rejection

7.0 DATE PERFORMED/TEST RESULTS 10-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-157

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TUR-7, Turbine DEH Computer Failure

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a turbine DEH computer failure.

3.0 TESTED OPTIONS

Failure of the P-2000 input to the DEH control system. Manual control is available.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 86% power.

5.0 TEST DESCRIPTION

The failure causes all automatic control functions to become inoperable. The DEH system automatically transfers to "Turbine Manual" control. The test is complete when the malfunction is cleared and the turbine is returned to automatic control, a smooth transition occurs and all indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-131.05, Digital Electrohydraulic Fluid System

7.0 DATE PERFORMED/TEST RESULTS 10-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 90-0862, DEH reference indication following controller reset

Minimal training impact. The controller response is as expected, however, the decimal points remain on when power is restored.

PERFORMANCE TEST ABSTRACT
MT-17

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSC-1, Refueling Water Storage Tank Leak

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a leak at the bottom of the refueling water storage tank. The leak rate can be selected from 0 to 10,000 gpm.

3.0 TESTED OPTIONS

The RWST develops a leak, located at the bottom of the tank, at a rate of 10,000 gpm.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all annunciators are verified and upon manual Safety Injection actuation and Containment Spray actuation, the RHR and Containment Spray pump suction valves properly align on RWST low-low level of 23.4 percent.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-110, Safety Injection System

6.3 OP-112, Containment Spray System

7.0 DATE PERFORMED/TEST RESULTS 10-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-21

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-1, Component Cooling Water Pump Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (8)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a CCW pump trip due to an overcurrent condition. Any combination of up to three CCW pumps may be selected simultaneously for the failure.

3.0 TESTED OPTIONS

Component Cooling Pump A trips due to overcurrent.

4.0 INITIAL CONDITIONS

Mode 1, 2 or 3, at normal temperature and pressure, with CCW Pump A in service.

5.0 TEST DESCRIPTION

The test will verify that on a CCW pump trip the standby CCW pump auto-starts and restores CCW flow and pressure to normal. All associated alarms will be verified to come in as expected. The test is complete when all annunciators that are required to clear on the start of the standby pump are verified clear.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-145, Component Cooling Water

6.3 AOP-014, Loss of Component Cooling Water

7.0 DATE PERFORMED/TEST RESULTS 4-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-210

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-27, Seal Water Heat Exchanger Tube Leak

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a seal water heat exchanger tube leak between the CVCS and CCW systems. The selected leak rate is a percent of system flow and will vary as system pressures vary.

3.0 TESTED OPTIONS

Seal Water Heat Exchanger tube leak.

4.0 INITIAL CONDITIONS

Mode 1, 2 or 3, at normal temperature and pressure, normal charging and letdown in service.

5.0 TEST DESCRIPTION

The test is complete when the indications of a seal water heat exchanger tube leak are verified and the leaking component is isolated by performing local operator actions from the instructor console to shut the heat exchanger inlet and outlet valves.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 10-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-22

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-2, Loss of CCW to RHR Heat Exchanger

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (7) and 3.1.2 (8)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a loss of CCW to an RHR heat exchanger due to the outlet valve failing shut. Any combination of the heat exchanger valves may be selected for the failure in the position of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

RHR heat exchanger B component cooling water outlet valve fails closed.

4.0 INITIAL CONDITIONS

Mode 4 or 5, with B RHR heat exchanger in service.

5.0 TEST DESCRIPTION

The test will be complete when 1CC-167 fails closed and the listed RCS parameters are verified to change in the correct direction, which will verify the loss of CCW.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-145, Component Cooling Water

6.3 AOP-014, Loss of Component Cooling Water

7.0 DATE PERFORMED/TEST RESULTS 1-18-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-23

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-3, CCW Leak into the Service Water System

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (8)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a component cooling water (CCW) leak from the CCW heat exchanger into the service water (SW) system. Due to the high flow rates of the SW and CCW systems, there is very little change in indicated flow and temperature. Any combination of CCW heat exchangers may be selected, with a leak rate in the range of 0 to 1000 gpm.

3.0 TESTED OPTIONS

CCW heat exchanger leak into SW system, 200 gpm.

4.0 INITIAL CONDITIONS

Mode 1, 2 or 3, at normal temperature and pressure, A CCW Pump in service.

5.0 TEST DESCRIPTION

The leak is detected by a decreasing CCW surge tank level and local operator action is taken to isolate the leaking heat exchanger. The test is complete when the CCW surge tank level is again stable and no longer decreasing, and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-145, Component Cooling Water System

6.3 AOP-014, Loss of Component Cooling Water

7.0 **DATE PERFORMED/TEST RESULTS** 6-28-90/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
MT-24

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-4, CCW Supply Valve Failure (Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (8)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a component cooling water (CCW) header supply valve failure. Either of the two supply header valves may be selected for the failure, to a position of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

CCW Nonessential header B supply valve failure, fully closed.

4.0 INITIAL CONDITIONS

Mode 1, 2 or 3, at normal temperature and pressure, B CCW Pump in service.

5.0 TEST DESCRIPTION

CCW system flows and pressures decrease and the standby CCW pump starts. The test is complete when system flow and pressure is returned to normal, and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-145, Component Cooling Water System

6.3 AOP-014, Loss of Component Cooling Water

7.0 DATE PERFORMED/TEST RESULTS 6-29-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-25

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-26, Letdown Heat Exchanger Tube Lek

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a letdown heat exchanger tube leak. Selected leak rate is from 0 to 300 gpm which is based on normal CVCS and CCW pressures. Direction of the leak and the leak rate are pressure dependent. If leakage is from the CVCS system, it will be limited by the letdown orifices in service.

3.0 TESTED OPTIONS

CVCS system develops a 300 gpm leak rate into the CCW system.

4.0 INITIAL CONDITIONS

Modes 1, 2 or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when indications and alarms of CVCS to CCW system in-leakage are verified along with increasing CCW activity levels and indication of letdown heat exchanger CCW relief valve lifting are observed.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical Volume Control System

6.3 OST-1007, CVCS/SI System Operability

6.4 OST-1106, CVCS/SI System Operability

6.5 OST-1825, SI ESF Response Time

6.6 OST-1826, SI ESF Response Time

7.0 DATE PERFORMED/TEST RESULTS 4-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-26

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-5, Loss of CCW to RCP Thermal Barrier

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (8)

2.0 AVAILABLE OPTIONS

The malfunction tests the response to a failure of FT-685, resulting in the closure of 1CC-252 and a loss of CCW to the RCP thermal barrier heat exchangers.

3.0 TESTED OPTIONS

RCP thermal barrier flow transmitter, FT-685, fails high and FCV-685 closes due to high flow indication.

4.0 INITIAL CONDITIONS

Mode 1, 2 or 3.

5.0 TEST DESCRIPTION

The test will be complete when the annunciator for thermal barrier high flow is received and 1CC-252 has been verified that it will stroke, but fails shut again when the control switch is released.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-145, Component Cooling Water

6.3 AOP-014, Loss of Component Cooling Water

7.0 DATE PERFORMED/TEST RESULTS 4-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-271

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-6, Letdown Temperature Controller Failure (TK-144, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the Letdown Temperature Controller, TK-144, while maintaining temperature in automatic.

3.0 TESTED OPTIONS

Letdown temperature controller fails closed.

4.0 INITIAL CONDITIONS

Modes 1, 2 or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test will verify that TCV-144 indicates shut and all associated alarms and auto actions occur in the letdown system, due to the increasing letdown temperature. The test will be complete when manual control of TCV-144 is verified and plant parameters respond as the valve is manually operated.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 4-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-272

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-6, Letdown Temperature Controller Failure (TK-144, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the Letdown Temperature Controller, TK-144. The valve can be selected to fail in auto from 0 to 100 percent, closed to open. Manual control is available.

3.0 TESTED OPTIONS

Letdown heat exchanger TK-144 fails high.

4.0 INITIAL CONDITIONS

Modes 1, 2 or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test will verify that TCV-144 indicates open and all associated alarms and auto actions occur in the letdown system, due to the decreasing letdown temperature. The test will be complete when manual control of TCV-144 is verified and plant parameters respond as the valve is manually operated.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 4-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-28

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CCW-7, Loss of CCW to the Reactor Coolant Pumps

1.2 ANSI/ANS 3.5, 1985 3.1.2 (8)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a loss of component cooling water to the reactor coolant pump oil coolers and thermal barrier heat exchangers.

3.0 TESTED OPTIONS

Component cooling water to RCP oil coolers and thermal barrier heat exchangers inlet valve fails closed.

4.0 INITIAL CONDITIONS

Modes 1, 2 or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test isolates CCW to the RCP oil coolers. The test is complete when the malfunction is entered and the RCP bearing temperatures increase due to the loss of CCW.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.3 AOP-018, Reactor Coolant Pump Abnormal Conditions

7.0 DATE PERFORMED/TEST RESULTS 4-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-31

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CWS-1, Circulating Water Pump Trip

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The procedure tests the proper response to the trip of a circulating water pump. Any of the three pumps may be selected for the failure.

3.0 TESTED OPTIONS

Circulating Water Pump 1B trip.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power, feedback loops out of service.

5.0 TEST DESCRIPTION

The turbine loses load as plant efficiency decreases due to a higher condenser back pressure and generator load decreases. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-138.01, Circulating Water

6.3 AOP-012, Partial Loss of Condenser Vacuum

7.0 DATE PERFORMED/TEST RESULTS 10-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-32

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CND-1, Main Condenser Tube Leak

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a main condenser tube leak. Either condenser zone may be selected with a leak rate of 0 to 1000 gpm.

3.0 TESTED OPTIONS

Tube leak of 1000 gpm in Main Condenser zone B.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Hotwell level increases to the condenser dump valve setpoint, which opens and rejects to the CST. CST level increases. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-033, Primary/Secondary Chemistry Out of Tolerance

7.0 DATE PERFORMED/TEST RESULTS 1-24-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-331

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CND-2, Hotwell Level Controller Failure (LC-1900, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (5)

2.0 AVAILABLE OPTIONS

The procedure tests proper response to the failure of the hotwell level controller failing high.

3.0 TESTED OPTIONS

Hotwell level controller failure, high.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Hotwell level decreases, condenser dump flow increases, the CST level increases, hotwell makeup level control valve opens and condenser makeup flow increases. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134, Condensate System

7.0 DATE PERFORMED/TEST RESULTS 6-29-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-333

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CND-2, Hotwell Level Controller Failure (LC-1901, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (5)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a failure of the hotwell level controller. Any combination of the level dump and makeup may be selected for the failure with a range of 0 to 6 feet. The malfunction does not affect the level transmitter, only the level controller output.

3.0 TESTED OPTIONS

Condenser makeup valve failure fully open, raising hotwell level until the reject valve opens.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The hotwell level increases to the hotwell dump setpoint and the dump valve opens to stop the level increase. The displacement of water to the hotwell is indicated by a level decrease in the CST. The test is complete when all indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134, Condensate System

7.0 DATE PERFORMED/TEST RESULTS 6-29-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-34

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CND-3, Loss of Condenser Vacuum

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (5)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a loss of condenser vacuum. The final value of condenser back pressure is selectable with a range of 0 to 30 psia.

3.0 TESTED OPTIONS

Loss of condenser vacuum to a final value of 30 psia.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power, with feedback loops out of service.

5.0 TEST DESCRIPTION

Load shed occurs as efficiency decreases until the turbine trips on low vacuum. The test is complete when the load shed actions, indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134, Condensate System

6.3 AOP-012, Partial Loss of Condenser Vacuum

7.0 DATE PERFORMED/TEST RESULTS 9-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-35

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CND-4, Loss of Condenser Vacuum Pump

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (5)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the loss of a condenser vacuum pump. Any combination of vacuum pumps may be selected to trip simultaneously, due to a high discharge temperature.

3.0 TESTED OPTIONS

Condenser vacuum pump "A" trip.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power, Vacuum Pump A in service.

5.0 TEST DESCRIPTION

The test is complete when the running vacuum pump trips and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-133, Main Condenser Air Removal System

6.3 OP-012, Partial Loss of Condenser Vacuum

6.4 2166-B-401, sheets 2101, 2101A, 2102, 2108

7.0 DATE PERFORMED/TEST RESULTS 12-5-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-41

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-1, Power Cabinet Urgent Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to an urgent failure in a rod control power cabinet. Any power cabinet may be selected for the failure. Rods in the unaffected power cabinets can be moved in the bank select mode.

3.0 TESTED OPTIONS

Power Cabinet 1AC urgent failure.

4.0 INITIAL CONDITIONS

Mode 3, with all rods inserted.

5.0 TEST DESCRIPTION

The test is complete when a ROD CONTROL URGENT FAILURE alarm is received for the rods in the 1AC Power Cabinet, the other rods are proven unaffected, and all other indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

6.3 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0267, urgent failure on C shutdown bank.

Minimal training impact as this problem only occurs in one specific condition. Instructor can plan the scenarios around it.

PERFORMANCE TEST ABSTRACT
MT-410

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-10, DRPI, Open or Shorted Coil

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a DRPI open or shorted coil. The failure may be selected individually or simultaneously for coil data circuits for each selectable first rod.

3.0 TESTED OPTIONS

Control Rod H-2 receives a fault in Data B circuit.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The procedure tests the proper response to the failure of a data coil for the selected rod. The test is complete when DRPI indication for the selected rod is verified to track only every 12 steps.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-411

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-11, Improper Bank Overlap

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The procedure tests the proper response to a mispositioning of the bank overlap thumb wheels, which causes an improper bank overlap, and three groups to attempt to move at the same time. Each control bank is selectable for the overlap failure with thumb wheel misposition settings selectable in a 0 to 612 step setting range.

3.0 TESTED OPTIONS

Overlap between Control Banks B and C has a thumb wheel setting of 220 steps which allows bank overlap misposition.

4.0 INITIAL CONDITIONS

Mode 3, with all shutdown banks withdrawn.

5.0 TEST DESCRIPTION

The procedure tests the proper response to a mispositioning of the overlap thumb wheels, which causes an improper bank overlap, and three groups of rods to attempt to move at the same time. The test is complete when a rod control urgent failure alarm is received and all rod motion stops.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-412

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-12, Control Bank Step Counter Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests for proper response to the failure of a step counter to move due to a data logging card failure. Each of the group step counter indicators may be selected for the failure.

3.0 TESTED OPTIONS

Group step indicator CD2 receives a failure of the bank's data logging.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

The test is complete when Control Bank D is inserted into the core, movement is verified by DRPI indication, and step counter CD2 has not changed position.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-413

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-13, Rod Speed Dead Band Control Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the dead band control circuitry of the rod speed program to a selected value. Failure of the rod control dead band program causes rods to control TAVE in the selected dead band. The dead band width may be selected with a range of 0 to 10°F.

3.0 TESTED OPTIONS

Rod control dead band circuitry fails to 10°F.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when a power increase is initiated and rod motion does not occur until TAVE is approximately 5° less than TREF.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 10-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-42

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-2, Logic Cabinet Urgent Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (13)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to an urgent failure in a rod control logic cabinet, which generates an URGENT FAILURE alarm. Rods will not move in any mode.

3.0 TESTED OPTIONS

Logic cabinet urgent failure.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power, with rod control in automatic.

5.0 TEST DESCRIPTION

The test is complete when a rod control URGENT FAILURE alarm is received, rods are proven to be immovable and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

6.3 AOP-001, Malfunction of Reactor Control System

7.0 **DATE PERFORMED/TEST RESULTS** 4-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-431

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-3, Dropped Rod

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a stationary gripper, resulting in one rod dropping. Any single rod can be selected for the malfunction with either the moveable or stationary gripper circuit selected with a circuit failure.

3.0 TESTED OPTIONS

Control Rod B6 dropped due to a failure in the stationary gripper circuit.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power, with rod control in manual.

5.0 TEST DESCRIPTION

The test is complete when the selected rod drops and the indications of temperature, power distribution, pressurizer pressure, and level are verified to respond correctly.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-44

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-4, Stuck Rod

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response for control rods being stuck, both trippable and nontrippable. Any rod may be selected for the malfunction with a choice of having one trippable or two nontrippable rods, and to allow two failures within the malfunction to be active at the same time.

3.0 TESTED OPTIONS

The plant is reducing turbine load and it is discovered that Control Rod H2 is stuck, untrippable, and Control Rod F6 is stuck, trippable.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

The test verifies the plant response to both a trippable and a nontrippable control rod. A reduction in power is commenced to cause a temperature error and rods to step in. The selected rods fail to insert with the other control rods, proving them to be stuck. The test is complete when a manual reactor trip is initiated to verify the rod selected as stuck, but trippable, inserts on the reactor trip.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-45

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-5, Ejected Rod

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an ejected rod. A shear of the CRDM housing, which caused a small LOCA, creates a selectable leak rate of 0 to 2000 gpm. Any rod may be designated for the malfunction.

3.0 TESTED OPTIONS

Rod H-14 is ejected creating an RCS leak of 2000 gpm. Depending on the position of Rod H-14 when ejected, positive reactivity effects may not be noticeable.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when the indications of an LOCA are verified as the result of the ejected rod and the indications of an ejected rod are verified on the DRPI panel.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-46

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-6, Uncontrolled Rod Motion

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of rod control which causes rods to continue stepping out at a constant rod speed after getting an initial rods out signal. Also tested is that rod motion stops when MANUAL is selected for the auto rod control failure. The failure may be selected to affect AUTO or MANUAL rod control and the speed may be selected to fail in a range of 8 to 72 steps per minute in the AUTO mode.

3.0 TESTED OPTIONS

Rod control system failure that allows uncontrolled rod motion in the out direction at 16 steps per minute in AUTO rod control and the MANUAL mode of rod control also fails to stop rod motion during the test.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

The procedure tests both the automatic and manual rod control malfunctions. After rod motion is in demand, the rods continue to step in the out direction. In automatic mode the rod speed is selectable, in manual the rods move at 48 spm. The test is complete when rod control is selected to MANUAL, an additional malfunction is inserted and rods step out with control still in MANUAL.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-47

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-7, Failure of Auto Rod Blocks to Block

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the auto rod control rod blocks to stop rod motion. During this test, the rod block that will be reached is the full rod withdrawal (C-11) interlock. Any of the six rod blocks may be selected for the failure. A malfunction in the rod control system defeats all automatic rod stop features.

3.0 TESTED OPTIONS

Rod control system Rod Block C-11, Control Bank D full rod withdrawal, will not function.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The procedure tests the proper response to the failure of Control Bank D full rod withdrawal Interlock C-11 to stop outward rod motion in automatic rod control. The test is complete when the rods reach 220 steps in automatic rod control and continue to step out to 228 steps.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-48

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-8, TREF Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (12)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the TREF processor to 580°F. The failed value is selectable in a range of 557 to 588°F.

3.0 TESTED OPTIONS

The TREF signal processor TY-408F fails to 580°F.

4.0 INITIAL CONDITIONS

Mode 1, at 100% power, with rod control in automatic.

5.0 TEST DESCRIPTION

The procedure tests the proper response to a failure of the TREF processor. The rods step in to lower TAVE to TREF. The lowering temperature causes steam pressure to decrease, pressurizer level and pressure to decrease, and a reduced megawatt output due to the reduced steam pressure. The test is complete when rod motion stops and TAVE is within the dead band of TREF.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-49

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CRF-9, DRPI Loss of Voltage

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to a loss of voltage to the DRPI coil groups. Coil groups may be selected individually or simultaneously.

3.0 TESTED OPTIONS

Voltage failure to DRPI Coil Groups A and B.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

6.3 AOP-001, Malfunction of Reactor Control System

7.0 DATE PERFORMED/TEST RESULTS 4-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0270, two or more rods at bottom annunciator.

Minor training impact as all the other alarms and indications did work as they should.

PERFORMANCE TEST ABSTRACT
MT-51

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-1, Letdown Isolation Valve Failure (1CS-11)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response of the letdown isolation valve failure due to a mechanical malfunction. The valve can be selected to fail in either the fully open or fully closed position without the capability of manual control.

3.0 TESTED OPTIONS

Letdown isolation valve fails fully closed.

4.0 INITIAL CONDITIONS

Mode 1, letdown in service.

5.0 TEST DESCRIPTION

The test is complete when the alarms and indications of 1CS-11 failing shut are verified and 1CS-11 will not open from the MCB.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5111

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-11, VCT Level Transmitter Failure (LT-112, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response of the VCT level transmitters failing. Either level transmitter may be selected for the failure. If LT-112 or LT-115 fails high, letdown will divert to the RHT and the high level alarm (for LT-115 only) will sound. If LT-115 fails low, the low level alarm will sound and automatic makeup will start.

3.0 TESTED OPTIONS

LT-112 fails high.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when all letdown diverts to the RHT and auto makeup to the VCT occurs at approximately 20 percent level.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5112

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-11, VCT Level Transmitter Failure (LT-115, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of LT-115. Either VCT level transmitter may be selected for the failure with a failed valve position of 0 to 100 percent.

3.0 TESTED OPTIONS

VCT level transmitter LT-115 fails low.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

VCT low level alarm will sound and automatic makeup will start. The test is complete when letdown diverts to the RHT as VCT level increases and level, as indicated on LI-115.1, indicates zero.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-512

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-14, RCP Number 1 Seal Failure (B RCP)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of an RCP Number 1 seal failure. Any of the three RCPs may be selected for the seal failure with a selectable leak rate of 30 to 177 gpm over a time period of 0 to 3600 seconds.

3.0 TESTED OPTIONS

B RCP Number 1 seal fails with a leak rate of 100 gpm.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all annunciators are verified and the Number 1 seal leakoff isolation valve is shut. The Number 2 seal will now control leakoff and RCS leakage will return to normal.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

6.3 AOP-018, Reactor Coolant Pump Abnormal Conditions

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-513

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-15, RCP Number 2 Seal Failure (A RCP)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a reactor coolant pump Number 2 seal failure on A RCP. Either A, B, or C RCP may be selected for the malfunction. As the Number 2 seal fails, leakoff causes a high flow alarm. A proportional decrease in the Number 1 seal leakoff may be observed.

3.0 TESTED OPTIONS

Number 2 seal fails on A RCP.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all annunciators are verified and the Number 2 seal leakage is confirmed by indications of the A RCP Number 1 seal leakoff decreasing on the Recorder Panel trace.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

6.3 AOP-018, Reactor Coolant Pump Abnormal Conditions

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-514

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 RCS-16, RCP Number 3 Seal Failure (C RCP)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to an RCP Number 3 seal failure. Any of the three RCPs may be selected for the malfunction. A rapid decrease in standpipe level and a slight containment sump level increase are observed.

3.0 TESTED OPTIONS

Number 3 seal fails on C RCP.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when all annunciators are verified and the leak is verified by indications of the automatic standpipe fill valve opening and containment sump level increasing due to the leak.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

6.3 AOP-018, Reactor Coolant Pump Abnormal Conditions

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5151

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-12, Boric Acid Flow Transmitter Failure (FT-113 to 20 GPM)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the boric acid flow transmitter to a selected value of 0 to 40 gpm.

3.0 TESTED OPTIONS

Boric acid flow transmitter, FT-113, fails to 20 gpm.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when the malfunction is entered and a boric acid flow deviation causes boration to stop. The boric acid flow recorder continues to read 20 gpm and the flow integrator continues to count.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5152

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-12, Boric Acid Flow Transmitter Failure (FT-113 to 0 gpm)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the boric acid flow transmitter, FT-113, to a selected value. The flow rate is selectable in a range of 0 to 40 gpm.

3.0 TESTED OPTIONS

Boric acid flow transmitter, FT-113, fails and boric acid flow goes to 0 gpm.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when the malfunction is entered and a boric acid flow deviation causes boration to stop. The boric acid flow recorder decreases to zero and total makeup flow indicates zero.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-516

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-13, Boric Acid Filter Plugged

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the boric acid filter becoming plugged. The blockage reduces the boric acid flow rate which may cause a flow deviation, if in the auto or borate modes. In the manual mode, boric acid flow is blocked allowing dilution to occur to the VCT. Blockage may be selected from a range of 0 to 100 percent.

3.0 TESTED OPTIONS

Boric acid filter blockage of 80 percent, with 20 gpm boration in progress and FK-113 selected to AUTO.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

FK-113 is selected to AUTO and a 20 gpm boration is started. The test is complete when the boric acid flow decreases to zero, the boric acid flow deviation causes makeup to secure, and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 4/19/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5181

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-15, Seal Injection Flow Control Valve Failure (HC-186, Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of seal injection flow control valve, HC-186. The failed position may be selected from 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Seal injection flow control valve, HC-186, fails fully open due to an I/P converter malfunction.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when seal injection flow, as indicated on the MCB, increases but stays less than 30 gpm total flow.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5182

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-15, Seal Injection Flow Control Valve Failure (HC-186, Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of seal injection flow control valve, HC-186. The failed position may be selected from 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Seal injection flow control valve, HC-186, fails fully closed due to an I/P converter malfunction.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when seal injection flow, as indicated on the MCB, decreases to zero, the seal injection low flow alarm is received and CCW temperatures, monitored on ERFIS, increase as a result of the loss of seal injection.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5201

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-17, Failure of Charging Flow Control Valve

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the response to a failure of FK-122, which causes FCV-122 to fail open. Selectable inputs are for valve failure from 0 to 100 percent, over a selectable ramp time of 0 to 3600 seconds.

3.0 TESTED OPTIONS

Controller fails FCV-122 to fully open position.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

FCV-122 failing open will cause VCT level to decrease and pressurizer level to increase. The test is complete when manual is selected on FK-122 and manual operation is verified to be available.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 4-20-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5202

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-17, Failure of Charging Flow Control Valve (Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of FK-122, which causes FCV-122 to shut, with manual control still available. The failed position is selectable with a range of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Charging Flow Control Valve (FCV-122) failure, fully closed.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when manual control is demonstrated available, CVCS parameters respond properly and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 1-4-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-523

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-20, High Temperature Divert Valve (TCV-143) Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of TE-143. This test also verifies that TCV-143 can be manually overridden to the DEMIN position. The failed temperature input to the divert valve can be selected from 50 to 200°F.

3.0 TESTED OPTIONS

High temperature divert valve temperature element, TE-143, fails to 200°F.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test will verify that, on a high temperature in the letdown line, TCV-143 will divert letdown flow around the demineralizers to protect the resin. The test is complete when the control switch for TCV-143 is placed in the DEMIN position to redirect letdown flow back to the demineralizers, thereby verifying that the divert signal can be overridden.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-20-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-524

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-21, Charging Pump Suction from RWST (LCV-115B or D) Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the charging pump suction valve from the RWST. Either valve may be selected for the failure with a value of 0 to 100 percent, closed to open. No manual control is available to the valve following the malfunction.

3.0 TESTED OPTIONS

Charging pump suction valve from RWST, LCV-115B, fails fully open.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

When the valve goes fully open, venting the VCT is commenced. This allows VCT pressure to decrease to less than RWST pressure. The test is complete when VCT level increases with indications of RWST to charging pump flow. RCS boron concentration is also observed to increase due to the boron concentration contained in the RWST.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-20-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-525

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-22, Charging Pump Miniflow Valve (1CS-182) Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the charging pump miniflow valve. Any of the three normal miniflow valves may be selected for the failure with a selectable valve position of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Charging pump miniflow valve, (1CS-182, fails fully closed.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure, with a charging pump in service.

5.0 TEST DESCRIPTION

The test will verify that when the miniflow valve fails closed, an increase in charging will be seen until the charging flow controller compensates for the increased flow. The test is complete when all alarms and indications are verified to be correct for the miniflow valve failure.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-20-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-526

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-23, Boric Acid Pump Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a trip of a boric acid pump during an automatic makeup. Either boric acid pump may be selected for the failure.

3.0 TESTED OPTIONS

Boric Acid Pump A trips due to an electrical fault.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure, with Boric Acid Pump A in AUTO and Boric Acid Pump B in OFF.

5.0 TEST DESCRIPTION

The test is complete when the selected boric acid pump trips and the various makeup system alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-20-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-527

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-24, Charging Line Containment Isolation Valve Failure

1.2 ANSI/ANS 3-5, 1985 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the charging line containment isolation valves. Either of the containment isolation valves may be selected for the failure of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Charging line containment isolation valve 1CS-235 fails fully closed.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test fails shut one of the charging line isolation valves, 1CS-235. This will stop normal charging flow to the RCS. Charging flow decreases and the low flow alarm is received. Charging flow control valve, FCV-122, will open, attempting to increase charging flow. Volume control tank level will increase and divert to the recycle holdup tank and pressurizer level will decrease to the letdown isolation setpoint of 17 percent. Pressurizer level then slowly increases.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-23-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5281

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-25A, Charging Line Leak of Charging Pump Suction

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a charging line leak. Location of the leak can be one of the following: at the pump suction, at the pump discharge, at the flow control valve, upstream of the regenerative heat exchanger or downstream of the regenerative heat exchanger, with a selectable leak rate of 0 to 200 gpm.

3.0 TESTED OPTIONS

A 150 gpm leak in the charging line at the charging pump suction, between the VCT and the charging pump suction header.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test verifies that VCT level will be lost and emergency makeup occurs at 5 percent level. This will isolate the VCT and place the charging pump suction on the RWST. The test is complete when RMS indicates increasing radiation in the RAB.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-16-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5282

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-25B, Charging Pump Discharge Line Leak before FT-122

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a charging line leak on the charging pump discharge header before FT-122. The leak may be selected for the following: at the charging pump suction or discharge, at FCV-122 or upstream or downstream of the regenerative heat exchanger. The leak rate is selectable from 0 to 200 gpm.

3.0 TESTED OPTIONS

Charging pump discharge line leak of 90 gpm, upstream of FT-122.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test verifies proper response to a leak at the charging pump discharge. VCT level decreases and charging flow decreases. FCV-122 opens to increase charging flow back to normal and stabilize pressurizer level. The test is complete when RMS indicates radiation levels increasing in the area of the leak.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-16-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5283

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-25C, Charging Line Leak between FT-122 and 1CS-235

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a charging line leak. Location of the leak can be one of the following: at the pump suction or discharge, at the flow control valve or upstream or downstream of the regenerative heat exchanger. The leak rate is selectable from 0 to 200 gpm.

3.0 TESTED OPTIONS

A 90 gpm leak in the charging line, downstream of the charging flow control valve.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test verifies the proper response to a leak downstream of the charging flow transmitter, causing a high charging flow to be sensed. The test is complete when charging flow increases to stabilize pressurizer level and radiation levels are verified to be increasing in the area of the leak.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-16-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5284

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-25, Charging Line Leak in Containment before Regen Hx

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b) (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a charging line leak. Location of the leak can be one of the following: at the pump suction or discharge, at the flow control valve or upstream or downstream of the regenerative heat exchanger, with a selectable leak rate of 0 to 200 gpm.

3.0 TESTED OPTIONS

A 90 gpm leak in the charging line, inside containment, upstream of the regenerative heat exchanger.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test verifies the proper response to pressurizer level and letdown parameters when a leak appears upstream of the regenerative heat exchanger. The test is complete when radiation levels inside containment start to increase and the leak is isolated by shutting charging isolation valves 1CS-235 and 1CS-238.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-5285

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-25, Charging Line Leak between Regen Hx and 1CS-492

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b) (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a charging line leak. Location of the leak can be one of the following: at the pump suction or discharge, at the flow control valve or upstream or downstream of the regenerative heat exchanger, with a selectable leak rate of 0 to 200 gpm.

3.0 TESTED OPTIONS

A 90 gpm leak in the charging line, downstream of the regenerative heat exchanger.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test verifies the proper response to pressurizer level and letdown parameters when a leak appears downstream of the regenerative heat exchanger. The test is complete when radiation levels inside containment start to increase and the leak is isolated by shutting charging isolation valves 1CS-235 and 1CS-238.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-53

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-3, Letdown Line Leak Inside Containment

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a letdown line leak inside containment, downstream of the orifice valves. The leak rate is selectable in percent of letdown flow, 0 to 100 percent (normal flow = 65 gpm).

3.0 TESTED OPTIONS

The in service letdown orifice develops a leak inside containment.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure, with one letdown orifice in service.

5.0 TEST DESCRIPTION

The leak will cause letdown pressure, flow and temperature to decrease and charging flow to increase to maintain pressurizer level. Containment radiation levels will increase. The test is complete when excess letdown is in service, normal letdown is isolated and the leak secured.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-016, Excessive Primary Plant Leakage

7.0 DATE PERFORMED/TEST RESULTS 4-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-54

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-4, Letdown Line Leak Outside Containment

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (1) (b)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a letdown line leak outside containment. The leak rate is in percent of letdown flow, 0 to 100 percent, for the in-service orifice.

3.0 TESTED OPTIONS

The letdown line develops a leak downstream of PCV-145, outside of containment.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure, with one orifice in service.

5.0 TEST DESCRIPTION

The leak comes in downstream of the letdown pressure control valve, PCV-145. A leak in this location will cause auto makeup control to cycle in order to maintain VCT level. No change will occur in pressurizer level since auto makeup control will restore any inventory lost from the system. The test is complete when radiation monitoring indication increases in the area of the leak.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-55

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-5, Charging Pump Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a charging pump trip due to an overcurrent relay actuation. Any of the three charging pumps may be selected for the malfunction.

3.0 TESTED OPTIONS

Charging Pump A trip on overcurrent.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure, and CSIP A in service.

5.0 TEST DESCRIPTION

CSIP A trips and indications of a loss of charging flow occur. Letdown temperatures increase, flashing occurs in the letdown line and pressurizer level and pressure decrease as a result of the loss of charging. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 9/5/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-56

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-6, Reactor Makeup Water Pump Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a reactor makeup water pump trip. Either pump may be selected for the trip.

3.0 TESTED OPTIONS

With Reactor Makeup Water Pump A running, Reactor Makeup Water Pump B control switch in STOP position, Reactor Makeup Water Pump A trip, due to an electrical fault.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The test is complete when the pump indicates tripped, and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 **DATE PERFORMED/TEST RESULTS** 9-5-90/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
MT-571

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-7, Letdown Pressure Control Valve Failure (PK-145, Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of PK-145 which causes PCV-145 stroke fully open. Manual control is still available. The valve position can be selected to fail 0 to 100 percent, fully closed to fully open.

3.0 TESTED OPTIONS

Letdown pressure control valve, PCV-145, fails fully open.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure, with one letdown orifice in service.

5.0 TEST DESCRIPTION

Failure of PCV-145 fully open will result in reduced pressure in the letdown line, a corresponding increase in letdown flow, a possible letdown line high flow alarm and/or high temperature divert, gradually increasing charging flow and possibly decreasing pressurizer level. Flashing may occur in the letdown line. The test is complete when PK-145 is placed in manual and pressure can be controlled.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-572

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-7, Letdown Pressure Control Valve Failure (PK-145, Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of PK-145, which causes PCV-145 to fail closed. Failure of the auto controller for PK-145 is selectable from 0 to 100 percent closed. Manual control of PK-145 is available from the MCB.

3.0 TESTED OPTIONS

Letdown Pressure Control Valve (PCV-145) failure, fully closed.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The demand signal to PK-145 fails to zero percent. Letdown pressure increases and letdown flow decreases. The test is complete when manual control of PK-145 is verified operable and all other indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 9-5-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-58

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-8, Loss of Normal Letdown

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a loss of normal letdown. Failure of bistable module 459C results in letdown isolation and loss of pressurizer heaters.

3.0 TESTED OPTIONS

Normal letdown is isolated due to failure of the pressurizer level bistable.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The malfunction causes letdown isolation valve LCV-459 to shut, isolating letdown. Letdown pressure and flow decrease to minimum and the pressurizer heaters deenergize on the interlock. The test is complete when the proper annunciators are verified to alarm and manual attempts to reopen LCV-459 are attempted, but the valve recloses when the switch is released.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 **DATE PERFORMED/TEST RESULTS** 4-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-59

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CVC-9, VCT Divert Valve (LCV-115A) Control Failure (RHT)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (18)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the auto control circuit for the VCT divert valve (LCV- 115A, causing the valve to fail to the recycle holdup tank (RHT). Also verified is the ability to take manual control of the valve. Failed valve position can be selected from full flow to the RHT or full flow to the VCT, 0 to 100 percent.

3.0 TESTED OPTIONS

The VCT divert valve (LCV-115A) controller fails, diverting full flow to the RHT.

4.0 INITIAL CONDITIONS

Mode 1, 2, or 3, at normal temperature and pressure.

5.0 TEST DESCRIPTION

The malfunction causes LCV-115A to indicate full flow to the RHT which results in decreasing VCT level to the auto makeup setpoint. The test is complete when LCV-115A is placed in manual and it indicates VCT position.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-107, Chemical and Volume Control System

7.0 DATE PERFORMED/TEST RESULTS 4-19-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-61

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-1, Station Blackout

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the response to a sequential tripping of breakers due to weather conditions, causing a station blackout. The blackout can be selected to occur with or without a delay.

3.0 TESTED OPTIONS

Station blackout.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The reactor trips due to loss of the reactor coolant pumps due to low frequency/voltage on the RCP buses. The emergency diesel generators start and pick up load to restore power to the emergency buses. The test is complete when the plant is stable, the proper equipment is verified to be in operation and all other indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-025, Loss of One Emergency AC Bus (6.9 kV) or Loss of One Emergency DC Bus (125V)

7.0 DATE PERFORMED/TEST RESULTS 9-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-610

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-8, Loss of a Unit Auxiliary Transformer (UAT)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to the loss of a unit auxiliary transformer due to a phase differential relay actuation. The lockout relay is actuated by the differential relay, tripping various breakers. Either A or B UAT is selectable for the malfunction.

3.0 TESTED OPTIONS

Unit Auxiliary Transformer phase differential relay trip.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The unit output breakers trip immediately, the exciter field breaker trips and a fast bus transfer is initiated to shift the in-house loads to the startup transformers. If the plant is above the P-7 Permissive setpoint, a reactor trip will occur following the turbine trip. No buses lose power with this malfunction. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-156.02, AC Electrical Distribution

7.0 DATE PERFORMED/TEST RESULTS 9-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0728, Relay response to loss of UAT.

Minimal training impact as all automatic actions occurred. The only problem was that one relay flag did not drop.

PERFORMANCE TEST ABSTRACT
MT-612

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GEN-3, Generator Output Breakers Fail to Trip

1.2 ANSI/ANS 3.5, 1985, N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a generator lockout, with a failure of the generator output breakers to open.

3.0 TESTED OPTIONS

Failure of Main Generator output breakers to open on Generator lockout actuation.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

A breaker failure scheme strips the switchyard buses to isolate the faulty generator output breakers. Power is maintained to plant loads via two switchyard breakers not directly connected to the switchyard buses. The test is complete when all actions, indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 1-22-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-615

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 DSG-2, Diesel Generator Governor Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a diesel generator's governor circuitry, while paralleled to it's associated emergency bus, running unloaded, and supplying it's associated emergency bus. Either diesel generator may be selected for the failure with the magnitude of oscillation in the range of 0 to 10 percent.

3.0 TESTED OPTIONS

Diesel Generator 1A-SA governor shunts:
- 1% and 10% unloaded
- 1% and 5% in parallel
- 5% while supplying emergency bus

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

Diesel generator load, current and frequency are observed, under the above stated conditions, as a result of the governor failure. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-155, Diesel Generator Emergency Power System

7.0 DATE PERFORMED/TEST RESULTS 1-5-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-616

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 DSG-3, Diesel Generator Breaker Inadvertent Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the trip of a diesel generator output breaker due to an inadvertent 86 lockout relay actuation.

3.0 TESTED OPTIONS

Diesel Generator A 86 lockout relay actuation.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the diesel generator trips and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-155, Diesel Generator Emergency Power System

7.0 DATE PERFORMED/TEST RESULTS 1-24-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0718, Trip annunciator not sealed in.

Minimal training impact. Diesel and plant respond as expected. Alarm actuates as expected. Alarm only fails to lock in.

PERFORMANCE TEST ABSTRACT
MT-617

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GEN-6, Generator Core Monitor Actuation (Insulation Failure)

1.2 ANSI/ANS 3.5, 1985, 3.1.2

2.0 AVAILABLE OPTIONS

This malfunction tests the proper response to a breakdown of the main generator stator winding insulation, resulting in the generator condition monitor actuating. The magnitude of the failure may be selected from 0 to 10 percent.

3.0 TESTED OPTIONS

Generator core monitor actuation with a failure magnitude of 10 percent.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when the ALARM status light is received on the generator core monitor panel, a computer alarm is generated on the MCB and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 OP-153.01, Generator, Exciter and Isolated Phase Bus System

6.2 AOP-006, Turbine Generator Trouble

6.3 APP-ALB-022, Annunciator Procedure

7.0 DATE PERFORMED/TEST RESULTS 9/11/90 SAT UNSAT

8.0 DEFICIENCIES FOUND DURING TESTING, CORRECTIVE ACTION TAKEN OR PLANNED AND ASSOCIATED DATES

SSR 90-0719, Generator malfunction for insulation failure.

9.0 EXCEPTIONS TAKEN AS A RESULT OF TEST PERFORMANCE, INCLUDING JUSTIFICATION

None

PERFORMANCE TEST ABSTRACT
MT-62

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-2, Loss of 120V AC Uninterruptable Power Supply (S-III)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the loss of a 120V AC uninterruptable power supply. Any of the four 120V AC instrument buses may be selected for the loss.

3.0 TESTED OPTIONS

Loss of 120V AC Instrument Bus S-III.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power, with PT-446 selected for turbine first stage pressure control.

5.0 TEST DESCRIPTION

Power is lost to the instrument bus with TREF failing low. Control Rods insert at 72 steps per minute until rods are placed in manual to stop rod motion. Main feedwater regulating valves are placed in manual to stop the increased feed flow and to regain control of SG levels. The test is complete when PT-447 is selected for turbine first stage pressure control, all controls are returned to automatic and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-024, Loss of Uninterruptable Power Supply

7.0 **DATE PERFORMED/TEST RESULTS** 9/6/90/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
MT-63

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-3, Loss of 125 VDC Emergency Bus (DP 1B-SB)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a loss of a 125 VDC emergency bus. Either 125 VDC emergency bus may be selected for the failure.

3.0 TESTED OPTIONS

A ground on 125 VDC Emergency Bus 1B-SB trips the battery charger and melts the connecting cable. (There are no breakers or fuses between the battery and bus.)

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when power is lost to the 125 VDC loads connected to the selected emergency DC bus, and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-025, Loss of Uninterruptable Power Supply

7.0 **DATE PERFORMED/TEST RESULTS** 9/6/90/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
MT-64

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-4, Loss of 6.9 kV Auxiliary Bus

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a loss of one of the 6.9 kV, nonemergency, auxiliary buses. Any of the five auxiliary buses, A through E, may be selected for the failure.

3.0 TESTED OPTIONS

Loss of 6.9 kV Auxiliary Bus B.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

One reactor coolant pump trips, resulting in a reactor trip. The test is complete when the trip occurs and all bus component indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-100, Reactor Coolant System

7.0 DATE PERFORMED/TEST RESULTS 9-7-90/SAT

8.0 DEFICIENCES FOUND DURING TESTING, CORRECTIVE ACTION TAKEN OR PLANNED AND ASSOCIATED DATES

None

PERFORMANCE TEST ABSTRACT
MT-65

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-5, Loss of a 6.9 Kv Emergency Bus

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests proper response to the trip of a 6.9 Kv feed breaker to an emergency bus. Any trip combination, up to two breakers to either emergency bus, may be selected. The associated emergency diesel generator starts and loads on the loss of the feed breaker. The second feed breaker is tripped to verify other interlocks requiring both sequencers to run Program A operate as required.

3.0 TESTED OPTIONS

Emergency Bus 1A-SA normal feed breaker trip. Emergency Bus 1B-SB feed breaker trip following Sequencer A completion.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The emergency bus loads deenergize on the loss of power. The emergency load sequencer runs and starts the required emergency safeguards equipment. Service water valve 1SW-276 strokes closed when Emergency Bus 1B-SB is deenergized. The test is complete when all expected actions occur and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-155, Diesel Generator Emergency Power System

6.3 OP-156.02, AC Electrical Distribution

6.4 AOP-025, Loss of One Emergency AC Bus or One Emergency DC Bus

7.0 DATE PERFORMED/TEST RESULTS 9-7-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-66

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-6, Loss of a 125 VDC Nonvital Bus (DP 1A)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the loss of a 125 VDC nonvital bus and other buses that may be powered from it. Any of the following buses may be selected for the failure: 250 VDC bus, 1A bus, 1A-1 bus and/or 1A-2 bus. Selection of 1A bus will also cause a loss of the 1A-1 and 1A-2 buses.

3.0 TESTED OPTIONS

DC Bus 1A supply breaker trip.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when loads being powered from Bus 1A are affected and all alarms and indications associated with the loss of Bus 1A are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 AOP-024, Loss of Uninterruptable Power Supply

7.0 DATE PERFORMED/TEST RESULTS 9/14/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-67

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 DSG-1, Diesel Generator Fail to Start

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3) and (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a diesel generator to start, due to a fuel rack control mechanism failure. Either diesel generator, or both, may be selected for the failure.

3.0 TESTED OPTIONS

Diesel Generator 1A-SA failure to start.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the diesel generator fails to start and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-155, Diesel Generator Emergency Power System

7.0 DATE PERFORMED/TEST RESULTS 9-7-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-68

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GEN-1, Automatic Voltage Regulator Failure (High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a main generator automatic voltage regulator failure. Regulator voltage will increase until the regulator transfers to manual. The failure of the regulator can be selected to a value of 0 to 20 percent.

3.0 TESTED OPTIONS

Main generator voltage regulator failure to 115 percent.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the voltage regulator transfers to manual control and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 9-7-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0699, Annunciator alarm.

Annunciator ALB-22-4-3. Generator volts to frequency ratio high or underfrequency relay did not alarm. Research is underway to determine if annunciator should alarm. No adverse training effects are expected.

PERFORMANCE TEST ABSTRACT
MT-69

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 EPS-7, Loss of Start-up Transformer

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (3)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the loss of a start-up transformer due to load side ground overcurrent. Either of the two start-up transformers may be selected for the failure.

3.0 TESTED OPTIONS

Start-up Transformer A lockout relay actuation on ground overcurrent.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

Auxiliary Buses 1A, 1C, 1D and Emergency Bus 1A-SA are lost. The emergency diesel generator starts, powers the emergency bus and the emergency sequencer runs in Program A, loss of off-site power. The test is complete when the actions have occurred and all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-156.02, AC Electrical Distribution

7.0 DATE PERFORMED/TEST RESULTS 9-10-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-710

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-9, Condensate Pump Trip (A Pump)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a condensate pump trip due to a bearing failure. Either condensate pump may be selected for the failure.

3.0 TESTED OPTIONS

Condensate Pump A trips due to an overcurrent condition.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 75 percent power.

5.0 TEST DESCRIPTION

The condensate pump trip results in a trip of its associated condensate booster pump and feedwater pump. The feedwater pump trip initiates a turbine runback and steam dumps actuate in response to the runback. The test is complete when the plant stabilizes at approximately 56 percent power and all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-25-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-711A

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-1, Motor-Driven Auxiliary Feed Pump Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a trip of an auxiliary feedwater pump. Any combination of the two motor-driven and/or one turbine-driven pumps may be selected for the failure. The motor-driven pumps trip on overcurrent and the turbine-driven pump trips on overspeed.

3.0 TESTED OPTIONS

Motor-driven AFW Pump 1A-SA overcurrent trip.

4.0 INITIAL CONDITIONS

Mode 3 with Auxiliary Feedwater Pump A maintaining SG levels.

5.0 TEST DESCRIPTION

The test is complete when the AFW pump trips, SG levels decrease, and all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-137, Auxiliary Feedwater System

6.3 SD-137, Auxiliary Feedwater System

7.0 DATE PERFORMED/TEST RESULTS 4-24-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-712

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CND-5, Failure of Excess Condensate Dump Valve, Closed

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (5)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of the condensate dump valve (LCV-1900) to the closed position.

3.0 TESTED OPTIONS

Condensate dump valve, LCV-1900, failure, closed.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power.

5.0 TEST DESCRIPTION

The test is complete when the hotwell level increases above the hotwell dump valve setpoint, the valve fails to open, and the hotwell level continues to increase to the high-level annunciator setpoint.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134, Condensate System

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-712A

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-1, Turbine-Driven Auxiliary Feed Pump Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a turbine-driven AFW pump tripping due to an overspeed condition.

3.0 TESTED OPTIONS

The turbine-driven AFW pump trips due to an overspeed condition.

4.0 INITIAL CONDITIONS

Mode 3 with the turbine-driven AFW pump maintaining SG levels.

5.0 TEST DESCRIPTION

The test is complete when the TDAFW pump trips, SG levels are decreasing, and all alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 **DATE PERFORMED/TEST RESULTS** 10-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-714

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSC-2, Condensate Storage Tank Leak

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a leak at the bottom of the condensate storage tank. The leak rate can be selected 0 to 3E7 lbm/hr.

3.0 TESTED OPTIONS

The condensate storage tank experiences a leak of 3E7 lbm/hr.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power.

5.0 TEST DESCRIPTION

The test is complete when the proper indications and alarms are verified as the condensate storage tank drains to empty.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 6-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-715

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-12, Heater Drain Pump Trip (B Pump)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a heater drain pump (HDP) trip due to overcurrent. Either pump may be selected for the failure.

3.0 TESTED OPTIONS

Heater Drain Pump B overcurrent trip.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power; feedback loops out of service.

5.0 TEST DESCRIPTION

Following the pump trip, main generator output decreases due to loss of plant efficiency. The test is complete when all indications and alarms are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-136, Feedwater Heaters, Vents, and Drains

7.0 DATE PERFORMED/TEST RESULTS 1-22-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-719

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-16, Main Feedwater Pump Trip (B Pump)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

Main feedwater pump trips due to overcurrent. Either Main Feedwater Pump A or B may be selected for the malfunction. This reduces feedwater flow to the steam generators with a reactor trip on low-low level if no operator action is taken.

3.0 TESTED OPTIONS

Main Feedwater Pump B trips on overcurrent. When reactor power stabilizes at approximately 56 percent, the malfunction is cleared and Main Feedwater Pump A is tripped on overcurrent, with Main Feedwater Pump B starting in automatic.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 85 percent power.

5.0 TEST DESCRIPTION

The procedure tests the proper response to a trip of one feedwater pump while at a high-power level. The turbine will run back to reduce steam flow to within the capacity of the remaining feedwater pump. After the plant is stable, the malfunction is cleared and the remaining feedwater pump trip is tripped to verify that the standby pump autostarts and restores feedwater flow.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134.01, Feedwater System

7.0 DATE PERFORMED/TEST RESULTS 4-30-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-72

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-2, Condensate Booster Pump Trip (B Pump)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

This tests the proper response to a condensate booster pump trip from 85 percent power. This will cause a turbine runback to approximately 345 psig first-stage pressure, which is approximately 56 percent reactor power. Either pump may be selected for the malfunction.

3.0 TESTED OPTIONS

Condensate booster pump trips due to an overcurrent condition.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 85 percent power.

5.0 TEST DESCRIPTION

The test verifies that on a loss of a booster pump, the corresponding main feedwater pump trips and a turbine runback occurs. The test is complete when all automatic actions and annunciators are verified and the plant is stable at approximately 56 percent power.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134, Condensate System

7.0 DATE PERFORMED/TEST RESULTS 10-16-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-720

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-17, Main Feedwater Pump Recirc Valve Failure (B Pump)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

The feedwater pump recirc valve modulation signal fails to a selected value. The recirc valve for either A or B main feedwater pump may be selected to fail from closed to fully open.

3.0 TESTED OPTIONS

Main Feedwater Pump B recirc valve fails fully open.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 90 percent power with the feedback loops in service.

5.0 TEST DESCRIPTION

The procedure tests the proper response to a main feedwater pump recirc valve failing open at power. The valve failing open will cause feedwater pressure to decrease and the steam generator levels will start to decrease. The feedwater regulating valves will respond and restore levels. The resultant decrease in plant efficiency will be demonstrated as reactor power increases to maintain the demanded megawatt output.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-30-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-721

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-18, Feedwater Flow Transmitter Failure (FT-486, Low)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9) (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a feedwater flow transmitter. Any of the feedwater flow transmitters are available for the failure with a failed value of 0 to 5E6 lbm/hr.

3.0 TESTED OPTIONS

Feedwater flow Transmitter FT-486 fails low to 2E6 lbm/hr.

4.0 INITIAL CONDITIONS

Mode 1 at 100 percent power, and FT-486 is selected for B SG level control.

5.0 TEST DESCRIPTION

The procedure tests the proper response to a controlling feedwater flow transmitter failing low which controls SG water level. The transmitter fails to a lower feedwater flow causing the level control system to react by raising SG level. The test is complete after all indications are verified, the failed channel is deselected for control, and the level is returning to normal.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-30-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-722

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-19, Feedwater Control Valve Position Failure LCV-488 Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9) (22)

2.0 AVAILABLE OPTIONS

The feedwater control valve fails to selected position due to a faulty valve controller. Manual control is available. Any of the three feedwater level control valves may be selected to fail from 0 to 100 percent closed to open.

3.0 TESTED OPTIONS

Steam Generator B feedwater level control valve fails fully open.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power.

5.0 TEST DESCRIPTION

The test is complete when B SG level increases to the level deviation alarm setpoint and manual control of the valve is verified to be operable.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-30-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-723

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-20, Feed Line Break Inside Containment

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a feed line break inside containment. Any of the three feedwater lines may be selected for the break with a variable leakage of 0 to 12E6 lbm/hr.

3.0 TESTED OPTIONS

Steam Generator C feedwater line breaks inside containment downstream of the feed check valve with a leak rate of 6E6 lbm/hr.

4.0 INITIAL CONDITIONS

Mode 1 approximately 100 percent power.

5.0 TEST DESCRIPTION

As C SG feed flow increases, the flow control valve will initially close and then reopen as C SG level decreases. The unaffected SG's feed flow decreases due to the break, while feed pressure continued to decrease and C SG level drops rapidly. The test is complete when the faulted SG continues to blow down into containment and there is 0 level indication in C SG.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-30-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-724

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-21, Feed Line Break Outside Containment

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a break in a main feed line outside containment. The leak can be selected for various locations: on C feed line between the flow detector and the FWIV, between the feed reg valve and the feed flow detector, the main feed header, and downstream of either main feed pump discharge and between their respective recirc valves. The leak can be selected to vary from 0 to 12E6 lbm/hr.

3.0 TESTED OPTIONS

Feed line C develops a leak of 6E6 lbm/hr between the feed flow detector and the FWIV.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power.

5.0 TEST DESCRIPTION

When the break occurs, feed flow to C SG increases rapidly while the flow to the other SGs decrease. The flow control valve to C SG closes due to the high flow. The test is complete when the affected SG level drops and the control valve opens again increasing the break flow. A reactor trip will occur on low-low SG level along with a start of the motor-driven AFW pumps.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-1-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-725

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SGN-1, Steam Generator Level Channel Failure

1.2 ANSI/ANS 3-5, 1985 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of an SG level channel low. Failures may be selected for any combination of the nine level transmitters simultaneously. Manual level control is possible. If failure is not on the controlling channel, only associated alarms and bistable lights will be received. Failure may be selected to a position of 0-100 percent full scale.

3.0 TESTED OPTIONS

Steam Generator C controlling Channel LT-496 fails low.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power.

5.0 TEST DESCRIPTION

The procedure tests the proper response to an SG level transmitter failing low. When the level transmitter fails low, the level control system will respond by increasing feed flow to the SG. The level will continue to increase until a turbine trip occurs on C SG high-high level.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-1-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-73

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-3, TD Auxiliary Feed Pump Speed Control Oscillates

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a turbine-driven AFW pump speed control circuit failure resulting in speed oscillations of a selected magnitude. Oscillation magnitude may be selected in the range of 0-1000 rpm.

3.0 TESTED OPTIONS

The turbine-driven auxiliary feedwater pump speed control oscillates at 500 rpm amplitude.

4.0 INITIAL CONDITIONS

Mode 3 with the turbine-driven AFW pump maintaining SG levels.

5.0 TEST DESCRIPTION

The turbine-driven AFW pump speed oscillates affecting the pump's discharge pressure and flow rate. This test is complete when manual control of the TDAFW pump is selected and the oscillation is verified to have stopped.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-24-90/SAT

8.0 DEFICIENCiES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-76

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-6, Auxiliary Feedwater Flow Control Valve Failure (Open)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of an AFW flow control valve to the fully open position. Any combination of the three control valves may be selected simultaneously with a selected failed position of 0 to 100 percent closed to open.

3.0 TESTED OPTIONS

An auxiliary feedwater flow control valve fails fully open.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 15 percent power with both MDAFW pumps running and throttled to provide 100 KPPH of flow to each SG.

5.0 TEST DESCRIPTION

The test is complete when C SG flow control valve fails open increasing flow to C SG and reducing flow to A and B SGs.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-25-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-771

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-7, Feedwater Bypass Valve Failure (Closed)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a main feedwater regulating bypass valve. Any of the feedwater bypass valves may be selected for the failure with a selected failed position of 0 to 100 percent closed to open.

3.0 TESTED OPTIONS

Feedwater regulating bypass valve to B SG, FCV-489, fails fully closed.

4.0 INITIAL CONDITIONS

Mode 1 with SG levels being maintained by the feedwater regulating bypass valves.

5.0 TEST DESCRIPTION

The test is complete when B SG feedwater regulating bypass valve fails shut and B SG level decreases to the level deviation alarm setpoint.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-25-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-772

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-7, Feedwater Bypass Valve Failure (Open)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (9)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a main feedwater regulating bypass valve. Either of the feedwater regulating bypass valves may be selected for the failure to a position of 0 to 100 percent, closed to open, if the valve is in automatic.

3.0 TESTED OPTIONS

Main feedwater regulating Valve B fails fully open.

4.0 INITIAL CONDITIONS

Mode 1 at approximately 100 percent power with SG levels being maintained by the feedwater regulating valves.

5.0 TEST DESCRIPTION

The test is complete when C SG feedwater regulating bypass valve fails open and C SG level increases to the point where C feedwater regulating valve receives a signal to close and maintain level.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-25-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-78

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 CFW-8, TDAFW Pump Flow Control Valve Failure (Shut)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a TDAFW flow control valve. Any of the flow control valves may be selected to fail from 0 to 100 percent closed to open.

3.0 TESTED OPTIONS

TDAFW FCV-2071A, to A SG, fails fully closed.

4.0 INITIAL CONDITIONS

Mode 3 with the TDAFW pump running.

5.0 TEST DESCRIPTION

The test is complete when AFW flow to A SG decreases to zero and flow to B and C SGs increase slightly with corresponding level increases.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 4-25-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-81

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-1, Steam Line Break Inside Containment

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a steam line break inside containment. Any of the steam lines may be selected for the failure with a leak rate of 0 to 12E6 lbm/hr.

3.0 TESTED OPTIONS

A steam line break inside containment with a leak rate of 2E6 lbm/hr.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

When the leak occurs, containment pressure increases to the safety injection actuation setpoint of 3 psig. A main steam isolation occurs and the faulted SG continues to blow down into containment until empty. All emergency safeguard actuations are verified by performing OMM-004 attachments.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OMM-004, Post Trip/Safeguards Review, Attachments 2-8, 10, 11, and Attachment 9 if containment spray actuation occurs

7.0 DATE PERFORMED/TEST RESULTS 10-23-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-810

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-7, Steam Dump Control Failure (Closed)

1.2 ANSI/ANS 3-5, 1985 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of the steam dump controller, which causes all steam dumps to fail closed. The selectable failure position of the valve is 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Steam dump valves fail fully closed, due to controller failure.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test verifies the proper response to the failure of the steam dump controller to open the steam dumps, following a load rejection. The test is complete when the plant is stable at approximately 75 percent power and the steam dumps are verified not to have opened during the transient.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-3-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-811

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-8, Mechanically stuck condenser dump valve

1.2 ANSI-3.5 Reference N/A

2.0 AVAILABLE OPTIONS

Any of the six condenser steam dumps, up to three simultaneously, may be stuck in any position from 0 to 100 percent.

3.0 TESTED OPTIONS

Failure of one steam dump valve stuck open 100 percent after steam dump actuation.

4.0 INITIAL CONDITIONS

Mode 1, 100% power

5.0 TEST DESCRIPTION

A 200 MW/minute ramp to 500 MW is initiated to arm C7A. Rods are held in manual until the second set of steam dumps open fully and then placed in auto. The steam dumps stabilize the plant and all steam dump valves reshut except the one stuck open. The steam dump mode selector switch is placed in RESET to reset C7A. The stuck valve still indicates open. The steam dump interlock switches are positioned to the OFF/RESET position and the stuck open valve remains open. The valve is verified shut when the malfunction is cleared.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-126, Main Steam and Steam Dump System

7.0 DATE PERFORMED/TEST RESULTS: 10-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-812

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-9, Steam Dump Permissive (P-12) Failure

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the response to a false Train A P-12 signal being generated, blocking normal steam dump operation.

3.0 TESTED OPTIONS

A false Train A P-12 signal is generated blocking normal steam dump operation.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test generates a false P-12 signal which blocks steam dump operation. The test is complete when a load rejection is initiated which requires steam dump operation, but the P-12 signal prevents the operation.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 **DATE PERFORMED/TEST RESULTS** 5-3-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-814

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-10, Steam Failure to TDAFW Pump (1MS-72 Closed)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (23)

2.0 AVAILABLE OPTIONS

The malfunction tests the response to a TDAFW pump steam supply valve failing shut. Either steam supply valve to the TDAFW pump can be selected.

3.0 TESTED OPTIONS

Main steam isolation valve to the TDAFW pump, 1MS-72, fails closed.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when a low-low level in two steam generators is generated, causing 1MS-70 to open and start the TDAFW pump, and 1MS-72 fails to open. The valve will not open from the MCB when the control switch is placed in the OPEN position.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 6-6-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-815

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Main Steam Header Break

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a steam line break downstream of the MSIVs. The leak size is selectable from 0 to 36E6 lbm/hr.

3.0 TESTED OPTIONS

Break in steam header just before the turbine throttle valves with a leak rate of 6E6 lbm/hr.

4.0 INITIAL CONDITIONS

Mode 1, approximately 85% power.

5.0 TEST DESCRIPTION

The test is complete following safety injection actuation and main steam isolation verification.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-3-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-82

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-2, Steam Line Break Outside Containment

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a steam line break outside containment. The procedure also verifies proper safeguards actuations by performing OMM-004, Attachments 2 through 8 and 10. The leak side is selectable with a leak rate of 0 to 12E6 lbm/hr.

3.0 TESTED OPTIONS

A steam line break outside containment on B steam line with a leak rate of 2E6 lbm/hr.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

When the steam line break is initiated, steam flow on all steam generators increase, RCS temperature drops, and pressurizer pressure and level decrease rapidly. Steam pressure drops causing safety injection and main steam line isolation. When the MSIVs close, steam flow from the non-affected SGs drops to zero with the affected SG's flow remaining high. Steam flow from the affected SG continues and the test is complete when it boils dry.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 10-24-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-83

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-3, Steam Header Pressure Detector PT-464 Failure, High

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a steam header pressure detector failing. The failed pressure range is selectable from 0 to 1300 psig.

3.0 TESTED OPTIONS

The steam header pressure detector PT-464 fails high at 1150 psig.

4.0 INITIAL CONDITIONS

Mode 3, with steam dumps in the steam pressure control mode.

5.0 TEST DESCRIPTION

The steam dumps will open, trying to decrease steam pressure to the setpoint value. The test is complete when temperatures decrease until the cooldown permissive P-12 causes the steam dumps to close.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-2-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-84

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-4, Steam Line Flow Transmitter FT-494 Failure, Low

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to Steam Generator C steam flow transmitter FT-494 failing. Any combination of up to six transmitters may be selected with a failed value of 0 to 5E6 lbm/hr.

3.0 TESTED OPTIONS

Steam line flow transmitter FT-494 fails low.

4.0 INITIAL CONDITIONS

Mode 1, with steam flow Channel FT-494 controlling C SG level.

5.0 TEST DESCRIPTION

The malfunction causes the feedwater regulating valve to close down, reducing feed flow and SG level. The test is complete when steam flow Channel FT-495 is selected to control SG level and the feedwater regulating valve reopens to stabilize the C SG at normal level.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-2-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-85

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SGN-2, Steam Generator Pressure Transmitter PT-485 Failure, High

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (22)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a steam generator pressure transmitter, high. Any combination of up to nine transmitters may be selected simultaneously with a failure range of 0 to 1300 psig.

3.0 TESTED OPTIONS

Steam Generator B pressure transmitter PT-485 fails on-scale high at 1200 psig.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power with steam flow Channel PT-485 selected for SG level control.

5.0 TEST DESCRIPTION

PI-485 fails high, resulting in an increase in steam flow indication and causing feedwater flow to increase. The resulting level increase may trip the turbine. The test is complete when the failed channel is deselected and the feedwater regulating valve receives a shut signal to match feed/steam flow and control B SG level.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-134.01, Feedwater System

7.0 DATE PERFORMED/TEST RESULTS 5-2-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-86

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SGN-3, Steam Generator Relief Valve Failure (A SG, Open)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a steam generator pressure transmitter, high, which results in the SG PORV failing open. Any of the SG PORVs may be selected with a selected failed value of 0 to 1300 psig.

3.0 TESTED OPTIONS

The pressure transmitter providing input to the A SG relief valve controller fails to 1300 psig.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

The SG PORV will fully open causing steam flow to increase which will cause reactor power to increase. The test is complete when manual control of the valve is demonstrated to be available.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-2-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-87

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-5, Main Steam Isolation Valve Failure (B SG, Shut)

1.2 ANSI/ANS 3.5, 1985 N/A

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a main steam isolation valve. The failure is selectable for either an inadvertent closure or a failure to close, with a ramp time of 0 to 3600 seconds.

3.0 TESTED OPTIONS

Inadvertent closure of B MSIV.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

Following the closure of B MSIV, pressure in B SG increases to the SG PORV setpoint. The PORV then controls pressure in the isolated SG. The test is complete when the plant is again stable and all alarms and indications are verified correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-126, Main Steam and Steam Dump System

7.0 DATE PERFORMED/TEST RESULTS 10-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-88

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 SGN-4, Steam Generator Safety Valve Failure (C SG, Open)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (20)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to the failure of a steam generator safety valve. Any of the safeties may be selected for the failure, with a selected failed position of 0 to 100 percent, closed to open.

3.0 TESTED OPTIONS

Steam Generator C safety valve fails fully open.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 50% power.

5.0 TEST DESCRIPTION

When the failure is initiated, steam flows from all of the SGs increase. RCS temperature decreases and reactor power increases. The test is complete when the reactor is manually tripped and the affected MSIV is closed. Indications of steam flow in the unaffected steam lines decrease to zero and the affected steam line continues to indicate steam flow.

6.0 BASELINE DATA/REFERENCES

Panel of experts

7.0 DATE PERFORMED/TEST RESULTS 5-2-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-89

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 MSS-6, Atomspheric steam dump valve failure

1.2 ANSI 3.5 Reference N/A

2.0 AVAILABLE OPTIONS

Three of the eight atmospheric steam dumps may be failed simultaneously from 0 to 100 percent.

3.0 TESTED OPTIONS

Failure of one atmospheric steam dump converter, causing one steam dump valve to fail open.

4.0 INITIAL CONDITIONS

Mode 1, 100% power

5.0 TEST DESCRIPTION

A malfunction for an atmospheric steam dump converter to fail is entered. A turbine load decrease of 999 MW/min is initiated to activate C7B. The affected steam dump valve fails open and all other steam dump valves operate normally. When the steam dump mode selector switch is placed in the RESET position, the affected steam dump valve reshuts and the test is terminated.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-126, Main Steam and Steam Dump System

7.0 DATE PERFORMED/TEST RESULTS: 10-11-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-91

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 NIS-1, Source Range Instrument Failure (N-31, High)

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a source range nuclear instrument channel failure in Mode 3. Either source range channel may be selected with the failure in a range of 5 to 1E6 cps.

3.0 TESTED OPTIONS

Source Range Channel N-31 failure, high, due to the pulse amplifier input from the detector.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

Source range counts increase, the start-up rate increases as the channel fails and the source range high flux and high flux at shutdown alarms annunciate. The test is complete when a reactor trip occurs and all indications and alarms for the failed channel are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-911

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 NIS-11, Source Range Instrument Power Fuse Blown

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a blown instrument power fuse in a source range drawer. Either or both of the channels may be selected with the instrument power, control power or both fuses chosen for the malfunction.

3.0 TESTED OPTIONS

Source Range Channel N-31 failure of an instrument power fuse in the source range drawer.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

When the instrument power fuse fails, the associated bistable lights are received and a reactor trip occurs. The test is complete when all alarms and indications are verified to be correct for the loss of instrument power to a source range drawer.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-912

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Intermediate Range Control Power Fuse Blown

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a blown control power fuse in an intermediate range drawer. Either of the two intermediate range channels may be may be selected with the instrument power, control power or both fuses selected for the malfunction.

3.0 TESTED OPTIONS

Intermediate range channel N-35 control power fuse blows.

4.0 INITIAL CONDITIONS

Mode 2, at 1E-8 amps.

5.0 TEST DESCRIPTION

When the control power fuse blows, a reactor trip occurs but no bistable lights on the selected intermediate range drawer are received. The test is complete when all alarms and indications are verified to be correct for the failure.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 5-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-913

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Power Range Control Power Fuse Blown

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a blown control power fuse in a power range channel. Any of the power range channels may be selected with either instrument power, control power or both fuses selected.

3.0 TESTED OPTIONS

Power range channel N-43 has a blown control power fuse in the power range drawer.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the alarms and indications for a blown control power fuse are received and verified correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 5-3-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-92

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 NIS-2, Source Range Pulse Height Discriminator Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a source range pulse height discriminator, resulting in an improper reference bias. Either source range channel may be selected with the failure in a range of 0 to 1E6 cps.

3.0 TESTED OPTIONS

Source Range Channel NI-31 pulse height discriminator failure, resulting in an improper reference bias.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

Source Range Channel N-31 indication increases, the audio count rate increases and a possible reactor trip occurs, depending on the initial count rate. The test is complete when all expected alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-93

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 NIS-3, Failure of Source Range Channel High Voltage to Disconnect

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a source range high voltage failure to deenergize, when blocked above P-6. Either or both of the source range channels may be selected for the failure.

3.0 TESTED OPTIONS

Source Range Channel N-32 voltage remains on after the block switches are activated.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The source range nuclear instruments are blocked above the P-6 setpoint, and N-32 fails to deenergize. Power is increased above the reactor trip setpoint of 1E5 cps and no trip occurs because it is blocked. The test is complete when all expected alarms and indications are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-94

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 NIS-4, Source Range Channel High Voltage Failure

1.2 ANSI/ANS 3.5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a source range high voltage power supply. Either source range channel may be selected with the failure in a range of 300 to 2500 volts.

3.0 TESTED OPTIONS

Source Range Channel NI-32 high voltage power supply failure.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when source range counts increase on N-32, the voltage indication on the source range drawer for N-32 increases to 2500 volts and the start-up rate pegs high, then decreases to zero as the failed channel's rate decays off.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 9-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-95

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Intermediate Range Channel Failure

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of an intermediate range channel due to a failure of the input to the log current amplifier. Any combination of up to two channels simultaneously may be selected for a failed value of 10E-11 to 10E-3 amps. A channel failure in the low direction will cause erroneous indication only. If failure is in the high direction, a possible rod stop and/or intermediate range high flux reactor trip could occur.

3.0 TESTED OPTIONS

Intermediate range channel N-35 fails high.

4.0 INITIAL CONDITIONS

Mode 2, at approximately 1E-8 amps.

5.0 TEST DESCRIPTION

The test is complete when a reactor trip occurs and the source range instruments fail to energize at 5E-11 amps as a result of the failed intermediate range channel.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 5-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
MT-96

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Intermediate Range Channel Gamma Compensation Failure

1.2 ANSI/ANS-3.5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of an intermediate range gamma compensation circuit. Either intermediate range channel may be selected with an additive current value selectable in a range of -1E-5 to 1E-5. If 5E-8 or greater is used, loss of compensation voltage alarm will activate.

3.0 TESTED OPTIONS

Intermediate range channel N-35 has a variation of the compensation channel setpoint of 10E-10.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DURATION

Following a manual reactor trip, NI-35 intermediate range level decreases to 1E-10 amps and levels off with the associated P-6 bistable not clearing. The test is complete when the source ranges are manually energized with the MCB reset switches and SUR indicates a negative trend.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Ex-Core Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 5-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-97

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Power Range Channel Detector Failure (Low)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a power range channel detector. Any combination of up to eight power range detectors may be selected with a final current of 0 to 5 milliamps. Normal full power is about .4 milliamp. Upper detectors are designated as A and lower detectors are designated as B.

3.0 TESTED OPTIONS

Power range channel N-44 Detector A fails low to 0 milliamps.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power with rods in automatic.

5.0 TEST DESCRIPTION

With N-44 detector failing low, Channel 4 axial flux difference indicates maximum negative deviation and rod speed demands jumps to 72 SPM with rod stop interlock for full Bank D withdrawal. All other indications and alarms are verified correct for the detector failure.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 5-3-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

MT-98

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Power Range Channel Failure (Low)

1.2 ANSI/ANS 3-5, 1985, 3.1.2 (21)

2.0 AVAILABLE OPTIONS

The malfunction tests the proper response to a failure of a power range channel summing amp. Any combination of up to four channels may be selected with a failed value of 0 to 200 percent.

3.0 TESTED OPTIONS

Power range channel N-44 fails low due to summing amplifiers output going to 0.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power with rods in automatic.

5.0 TEST DESCRIPTION

The failure results in erroneous indications and various protection bistable indications and alarms but does not generate a reactor trip since only one channel will be affected. Rod speed demand jumps to 72 SPM then decreases to 0 SPM as the rate circuit dies off and TAVE/TREF mismatch increases. The test is complete when all indications and alarms are verified to be correct for the power range low failure.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-105, Excore Nuclear Instrumentation

7.0 DATE PERFORMED/TEST RESULTS 5-3-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

OST-1004

NOST-001

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1004, Power Range Heat Balance

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a power range heat balance during normal plant operations.

3.0 TESTED OPTIONS

Power range heat balance performed with normal operating conditions.

4.0 INITIAL CONDITIONS

Mode 1, equilibrium xenon conditions.

5.0 TEST DESCRIPTION

The test is complete when all data collection, calculations and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1004, Power Range Heat Balance

6.3 SHNPP, Cycle 3, EXPACK, Version 2.0

7.0 DATE PERFORMED/TEST RESULTS 9/11/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 90-0711, RCP heat input appears high

SSR written to tune the simulator. OST was satisfactorily performed. No training impact.

8.2 SSR 90-0723, NI power levels at 100% power

SSR was written to reshoot the IC to capture different NI gains. No training impact.

PERFORMANCE TEST ABSTRACT
OST-1007
NOST-002

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1007, CVCS/SI System Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to verify the CVCS/SI System operability during normal plant operations.

3.0 TESTED OPTIONS

System operability verification with normal operating conditions.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1007, CVCS/SI System Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9/12/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1008
NOST-003

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1008, 1A-SA RHR Pump Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the RHR pump operability surveillance test during normal plant operations, Modes 1, 2 or 3.

3.0 TESTED OPTIONS

RHR pump operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1008, 1A-SA RHR Pump Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 1-18-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1009
NOST-004

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1009, Containment Spray Pump Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the Containment Spray Pump operability surveillance test during normal plant operations, Modes 1, 2, 3 or 4.

3.0 TESTED OPTIONS

Containment Spray pump operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1009, Containment Spray Pump Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1013
NOST-005

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1013, 1A-SA Emergency Diesel Generator Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the emergency diesel generator operability surveillance test during normal plant operations, Modes 1 through 6.

3.0 TESTED OPTIONS

Emergency diesel generator operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1013, 1A-SA Emergency Diesel Generator Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9/12/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1018
NOST-007

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1018, Main Steam Isolation Valve and Main Feedwater Isolation Valve Operability Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the main steam and main feedwater isolation valve operability surveillance test during normal plant operations.

3.0 TESTED OPTIONS

Main steam and main feedwater isolation valve operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.1 OST-10184, Main Steam Isolation Valve and Main Feedwater Isolation Valve Operability Test

7.0 DATE PERFORMED/TEST RESULTS 9-12-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1021
NOST-008

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1021, Daily Surveillance Requirements

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the daily surveillance requirements during normal plant operations, Modes 1 or 2.

3.0 TESTED OPTIONS

Daily surveillance requirements, rod positions, acceptable range and setpoint verification and channel checks, performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.1 OST-1021, Daily Surveillance Requirements

7.0 DATE PERFORMED/TEST RESULTS 9/13/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0707, MIMS not modeled.

Minimal training impact. Training value assessment requested by SRG meeting 11/16/90.

PERFORMANCE TEST ABSTRACT
OST-1022
NOST-009

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1022, Daily Surveillance Requirements

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the daily surveillance requirements during normal plant operations, Mode 3.

3.0 TESTED OPTIONS

Daily surveillance requirements, acceptable range and setpoint verification and channel checks, performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1022, Daily Surveillance Requirements

7.0 DATE PERFORMED/TEST RESULTS 9/13/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1026
NOST-010

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1026, Reactor Coolant System Leakage Evaluation Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a reactor coolant system leakage evaluation surveillance test during normal plant operations, Modes 1 through 4.

3.0 TESTED OPTIONS

Reactor coolant system leakage evaluation surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection, calculations and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1026, Reactor Coolant System Leakage Evaluation Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9/13/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1036
NOST-011

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1036, Shutdown Margin Calculation Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a shutdown margin calculation surveillance test during normal plant operations, Modes 1 through 6.

3.0 TESTED OPTIONS

Shutdown Margin Calculation surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection, calculations and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1036, Shutdown Margin Calculation Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-14-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1039
NOST-012

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1039, Quadrant Power Tilt Ratio Calculation Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a quadrant power tilt ratio calculation surveillance test during normal plant operations.

3.0 TESTED OPTIONS

Quadrant power tilt ratio calculation surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection, calculations and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1039, Quadrant Power Tilt Ratio Calculation Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1046
NOST-013

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1046, Main Steam Isolation Valve Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a main steam isolation valve operability surveillance test during normal plant operations.

3.0 TESTED OPTIONS

Main steam isolation valve operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1046, Main Steam Isolation Valve Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1054
NOST-014

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1054, Permissives P-6 and P-10 Verification Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a permissives P-6 and P-10 verification surveillance test during normal plant operations, Modes 3 through 6.

3.0 TESTED OPTIONS

Permissives P-6 nd P-10 verification surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1054, Permissives P-6 and P-10 Verification Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1073
NOST-015

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1073, 1B-SB Emergency Diesel Generator Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the emergency diesel generator operability surveillance test during normal plant operations, Modes 1 through 6.

3.0 TESTED OPTIONS

Emergency diesel generator operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1073, 1B-SB Emergency Diesel Generator Operability Surveillance Test

7.0 **DATE PERFORMED/TEST RESULTS** 9/14/90/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

None

PERFORMANCE TEST ABSTRACT
OST-1075
NOST-016

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1075, Turbine Mechanical Overspeed Trip Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a main turbine mechanical overspeed trip surveillance test during normal plant operations, Modes 1 and 2.

3.0 TESTED OPTIONS

Main turbine mechanical overspeed trip surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 2.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1075, Turbine Mechanical Overspeed Trip Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1076
NOST-017

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1076, Auxiliary Feedwater Pump 1B-SB Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the Auxiliary Feedwater Pump 1B-SB operability surveillance test during normal plant operations, Modes 1 through 4.

3.0 TESTED OPTIONS

Auxiliary Feedwater Pump 1B-SB operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1076, Auxiliary Feedwater Pump 1B-SB Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9/17/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0772, Recirc flow unsatisfactory

Minimal training impact as the rest of the surveillance can be done satisfactorily.

PERFORMANCE TEST ABSTRACT
OST-1092
NOST-018

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1092, 1B-SB RHR Pump Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the 1B-SB RHR pump operability surveillance test during normal plant operations, Modes 1, 2 or 3.

3.0 TESTED OPTIONS

1B-SB RHR pump operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1092, 1B-SB RHR Pump Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 1-18-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1126
NOST-019

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1126, Reactor Coolant Pump Seals Controlled Leakage Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the reactor coolant pump seals controlled leakage surveillance test during normal plant operations, Modes 1 through 4.

3.0 TESTED OPTIONS

Reactor coolant pump seals controlled leakage surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1126, Reactor Coolant Pump Seals Controlled Leakage Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1211
NOST-020

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1076, Auxiliary Feedwater Pump 1A-SA Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform the Auxiliary Feedwater Pump 1A-SA operability surveillance test during normal plant operations, Modes 1 through 4.

3.0 TESTED OPTIONS

Auxiliary Feedwater Pump 1A-SA operability surveillance test performed in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test procedure and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1211, Auxiliary Feedwater Pump 1A-SA Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 9/17/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0772, Recirc flow unsatisfactory

Minimal training impact as the rest of the surveillance can be done satisfactorily.

PERFORMANCE TEST ABSTRACT
OST-1316
NOST-021

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1316, Component Cooling Water System Operability Surveillance Test

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to verify the CCW System operability during normal plant operations, any mode, 1 through 4.

3.0 TESTED OPTIONS

System operability verification with normal operating conditions.

4.0 INITIAL CONDITIONS

Mode 1.

5.0 TEST DESCRIPTION

The test is complete when all data collection and procedural steps are performed in accordance with the surveillance test and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OST-1316, Component Cooling Water System Operability Surveillance Test

7.0 DATE PERFORMED/TEST RESULTS 1-21-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
OST-1005
NOST-023

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 OST-1005, Control Rod and Rod Position Indicator Exercise

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (10)

2.0 AVAILABLE OPTIONS

The surveillance procedure tests the response during the monthly control rod and rod position indicator exercise.

3.0 TESTED OPTIONS

The exercise is performed in accordance with the surveillance procedure.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 90% power.

5.0 TEST DESCRIPTION

The test is complete when it is verified that each rod will move at least 10 steps in any one direction, as indicated by DRPI, and the individual rod positions, as indicated by DRPI, are in agreement with the step counters, within ± 12 steps.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 OP-104, Rod Control System

7.0 DATE PERFORMED/TEST RESULTS 10-25-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
GP-006
NOT-001

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GP-006, Normal Plant Shutdown from Power Operation to Hot Standby

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (8)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a normal plant shutdown from power operation to a hot standby condition.

3.0 TESTED OPTIONS

Plant shutdown in accordance with approved plant procedures, Mode 1 to Mode 3.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when a normal plant shutdown is performed, normal operating temperatures and pressures are maintained, all data collection and procedural steps are performed in accordance with controlled procedures and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 GP-006, Normal Plant Shutdown from Power Operation to Hot Standby

7.0 DATE PERFORMED/TEST RESULTS 9-18-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 90-0780, BTRS not functional

System hasn't been used in 3 years in the plant but it is scheduled to become operational in the simulator by 1-31-92.

8.2 SSR 90-0779, NIS channel deviation

Minimal training impact as operator can use NI indication to evaluate actual deviation.

8.3 SSR 90-0760, Hotwell level inconsistencies

Minimal training impact as variations are small enough not to be noticed by trainee.

PERFORMANCE TEST ABSTRACT
GP-007
NOT-002

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GP-007, Normal Plant Cooldown

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (8)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a normal plant cooldown from hot standby to a cold shutdown condition.

3.0 TESTED OPTIONS

Plant cooldown in accordance with approved plant procedures, Mode 3 to Mode 5.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when a normal plant cooldown is performed, normal operating temperatures and pressures are maintained, all data collection and procedural steps are performed in accordance with controlled procedures and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 GP-007, Normal Plant Cooldown

7.0 DATE PERFORMED/TEST RESULTS 9-24-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 90-0818, Seal injection with CSIP off.

VCT float outside scope of simulation per SRG assessment.

8.2 SSR 90-0816, Liquid/vapor temperature mismatch

This mismatch was seen one time during an attempt to draw a bubble. Other bubble evolutions were satisfactory. SSR outstanding to attempt to reproduce the problem.

PERFORMANCE TEST ABSTRACT
GP-002
NOT-003

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GP-002, Normal Plant Heatup from Cold Solid to Hot Subcritical

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (1)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a normal plant heatup from a cold, solid condition to a hot, subcritical condition.

3.0 TESTED OPTIONS

Plant heatup in accordance with approved plant procedures, Mode 5 to Mode 3.

4.0 INITIAL CONDITIONS

Mode 5, solid plant conditions.

5.0 TEST DESCRIPTION

The test is complete when a normal plant heatup is performed, normal operating temperatures and pressures are maintained, all data collection and procedural steps are performed in accordance with controlled procedures and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 GP-002, Normal Plant Heatup from Cold Solid to Hot Subcritical

7.0 DATE PERFORMED/TEST RESULTS 9/27/90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 90-0804, PRZ vapor/liquid space temperatures not equal during bubble formation

Problem seen only once. PRZ bubble can be drawn per procedure.

8.2 SSR 90-0787, SR/IR readings did not increase during heatup

Minimal training impact as these indications have no effect on heatup

8.3 SSR 90-0772, MDAFW pump recirc flow too low

Minimal impact as value is only slightly too low

PERFORMANCE TEST ABSTRACT
GP-004
NOT-004

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GP-004, Reactor Startup

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (2)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform a normal reactor startup.

3.0 TESTED OPTIONS

Reactor startup performed (Mode 3 to Mode 2) in accordance with approved plant procedures.

4.0 INITIAL CONDITIONS

Mode 3.

5.0 TEST DESCRIPTION

The test is complete when a normal reactor startup is performed using controlled procedures and all indications and responses are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 GP-004, Reactor Startup

7.0 DATE PERFORMED/TEST RESULTS 9-27-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
GP-005
NOT-005

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 GP-005, Power Operations

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (2), (3), (6)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to perform normal plant power operations.

3.0 TESTED OPTIONS

Power operations in accordance with approved plant procedures, Mode 2 to Mode 1.

4.0 INITIAL CONDITIONS

Mode 2.

5.0 TEST DESCRIPTION

A unit startup to rated power is performed using controlled plant procedures. The test is complete when the plant is at rated power, normal operating temperatures and pressures are maintained, all data collection and procedural steps are performed in accordance with controlled procedures and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 GP-005, Power Operation

7.0 DATE PERFORMED/TEST RESULTS 10-4-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 90-0842, MW recorder readings not realistic when loads shifted to UAT.

Minimal impact as MW recorders are only off about 20 MW and operators don't rely on recorders for this evolution.

8.2 SSR 90-0830, ERFIS points on Circ Water

Minimal training impact due to backup indication being seldom used.

8.3 SSR 90-0833, Turbine Eccentricity and Vibration < 600 rpm.

Minimal training impact as problem only occurs with turbine less than 600 rpm.

8.4 SSR 90-0839, Turbine Aux. Coolers

Presently no local throttle capability for coolers; being evaluated for addition. System functions correctly.

PERFORMANCE TEST ABSTRACT

Recovery to Rated Power Following Reactor Trip

NOT-006

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Recovery to Rated Power following a Reactor Trip

1.2 ANSI/ANS 3.5, 1985, 3.1.1 (4)

2.0 AVAILABLE OPTIONS

The procedure tests the ability to recover from a reactor trip and return to rated power operation.

3.0 TESTED OPTIONS

Reactor trip recovery and return to power operation in accordance with approved plant procedures, Mode 3 to Mode 1.

4.0 INITIAL CONDITIONS

Mode 1, approximately 100% power, equilibrium conditions.

5.0 TEST DESCRIPTION

Following a reactor trip, the unit is restored to power operation in accordance with controlled plant procedures. The test is complete when the plant is at rated power, normal operating temperatures and pressures are maintained, all data collection and procedural steps are performed in accordance with controlled procedures and are verified to be correct.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts

6.2 Path-1

6.3 EOP-EPP-004, Reactor Trip Response

6.4 GP-006, Normal Plant Shutdown from Power Operation to Hot Standby (Mode 1 to Mode 3)

6.5 GP-004, Reactor Startup (Mode 3 to Mode 2)

6.6 GP-005, Power Operation (Mode 2 to Mode 1)

7.0 DATE PERFORMED/TEST RESULTS 10-9-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

SSR 90-0842, MW recorder readings not realistic when loads shifted to UAT.

Minimal impact as MW recorders are only off about 20 MW and operators don't rely on recorders for this evolution.

PERFORMANCE TEST ABSTRACT
RTT-001

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Simulator Real Time-Test

1.2 ANSI/ANS 3.5, 1985, 3.1.2

2.0 AVAILABLE OPTIONS

The real-time test ensures that the simulator is capable of performing required plant evolutions, normal and emergency, and required malfunctions/transients in real time.

3.0 TESTED OPTIONS

Spare time routine (SPARDUTY) to collect data.

4.0 INITIAL CONDITIONS

Mode 1, at approximately 100% power.

5.0 TEST DESCRIPTION

The test is complete when the simulator is run for a specified duration, the worst case data is recorded, and a subjective evaluation of any hesitation, if any, on the impact to the student is made. The specific test runs for 5 minutes at a steady state of approximately 100 percent power, a severe LOCA transient conditions is initiated, the simulator is run for an additional 10 minutes, and data is collected. The test is then repeated in order to observe any pauses/hesitations that may have occurred and make an evaluation as to their impact on the students.

6.0 BASELINE DATA/REFERENCES

Processor spare times are greater than zero.

7.0 **DATE PERFORMED/TEST RESULTS** 1-18-91/SAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

SSR 91-0080, simulator real-time test.

Minor pauses/hesitation noted in annunciators will not result in negative training and will not distract operator's attention from normal activities.

PERFORMANCE TEST ABSTRACT
SST-001

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Steady-State Accuracy and Stability Test, 100% Power

1.2 ANSI/ANS-3.5, 1985, 4.1 and A3.2

1.3 Regulatory Guide 1.149, Rev. 1, April 1987

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

3.1 Demonstration of simulator steady-state accuracy for one hour during power operation.

3.2 Verification of control board indications to actual plant values.

4.0 INITIAL CONDITIONS

Mode 1, 100% power, BOL, equilibrium Xenon.

5.0 TEST DESCRIPTION

5.1 Stability Test

After the simulator has been initialized, the parameter monitoring program is run for 60 minutes for the appendix B2.1 parameters. The program will report any parameters which do not stay within a 2% band of their initial values.

5.2 Accuracy Test

The simulator is operated in accordance with approved plant procedures until it reaches the conditions of the reference plant, 100% power, log readings. The simulator data is then recorded. The simulator is then frozen and a comparison is made between the reference plant and simulator readings.

6.0 BASELINE DATA/REFERENCES

Plant 100% power data of 12-6-88.

7.0 DATE PERFORMED/TEST RESULTS: 1-19-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 91-0072, intermediate range NIs are reading slightly low in the simulator.

Minimal training impact as IRs are not used in the power range for exact indication of core power.

8.2 SSR 91-0073, power range delta fluxes are lower (more negative) in the simulator than in the plant.

Minimal training impact as the simulator values are still in the correct band and they do vary with plant operations and malfunctions.

8.3 SSR 91-0074, CCW heat exchanger outlet temperature was 16 degrees higher in the simulator than in the plant.

Minimal training impact as all values were in the normal range.

8.4 SSR 91-0075, power range nuclear instrument upper and lower detector currents on the recorder panel were too low in the simulator.

Minimal training impact. This will be corrected as a deficiency, however, this indication is infrequently used by the operators in simulator training. Meter indication of power was correct.

PERFORMANCE TEST ABSTRACT
SST-002

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Steady-State Accuracy Test, 75% Power

1.2 ANSI/ANS-3.5, 1985, 4.1 and A3.2

1.3 Regulatory Guide 1.149, Rev. l, April 1987

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Verification of control board indications to actual plant values.

4.0 INITIAL CONDITIONS

Mode 1, 75% power, BOL.

5.0 TEST DESCRIPTION

The simulator is operated in accordance with approved plant procedures until it reaches the conditions of the reference plant, 75% power, log readings. The simulator data is then recorded. The simulator is then frozen and a comparison is made between the reference plant and simulator readings.

6.0 BASELINE DATA/REFERENCES

Plant 75% power data of 10-27-88.

7.0 **DATE PERFORMED/TEST RESULTS:** 1-20-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING TRAINING IMPACT

8.1 SSR 91-0071, turbine impulse pressure reading slightly higher in the simulator than in the plant.

Minimal training impact as the value is only about 25 psig high and megawatts are comparable.

8.2 SSR 91-0072, intermediate range NIs are reading slightly low in the simulator.

Minimal training impact as IRs are not used in the power range for exact indication of core power.

8.3 SSR 91-0073, power range delta fluxes are lower (more negative) in the simulator than in the plant.

Minimal training impact as the simulator values are still in the correct band and they do vary with plant operations and malfunctions.

8.4 SSR 91-0074, CCW heat exchanger outlet temperature was 16 degrees higher in the simulator than in the plant.

Minimal training impact as all values were in the normal range.

8.5 SSR 91-0075, power range nuclear instrument upper and lower detector currents on the recorder panel were too low in the simulator.

Minimal training impact. This will be corrected as a deficiency, however, this indication is infrequently used by the operators in simulator training. Meter indication of power was correct.

PERFORMANCE TEST ABSTRACT
SST-003

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 Steady-State Accuracy Test, 30% Power

1.2 ANSI/ANS-3.5, 1985, 4.1 and A3.2

1.3 Regulatory Guide 1.149, Rev. 1, April 1987

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Verification of control board indications to actual plant values.

4.0 INITIAL CONDITIONS

Mode 1, 30% power, BOL.

5.0 TEST DESCRIPTION

The simulator is operated in accordance with approved plant procedures until it reaches the conditions of the reference plant, 30% power, log readings. The simulator data is then recorded. The simulator is then frozen and a comparison is made between the reference plant and simulator readings.

6.0 BASELINE DATA/REFERENCES

Plant 30% power data of 10-25-88.

7.0 DATE PERFORMED/TEST RESULTS: 1-20-91/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

8.1 SSR 91-0073, power range delta fluxes are lower (more negative) in the simulator than in the plant.

Minimal training impact as the simulator values are still in the correct band and they do vary with plant operations and malfunctions.

8.2 SSR 91-0074, CCW heat exchanger outlet temperature was 16 degrees higher in the simulator than in the plant.

Minimal training impact as all values were in the normal range.

8.3 SSR 91-0075, power range nuclear instrument upper and lower detector currents on the recorder panel were too low in the simulator.

Minimal training impact. This will be corrected as a deficiency, however, this indication is infrequently used by the operators in simulator training. Meter indication of power was correct.

8.4 SSR 91-0077, condenser vacuum too high in the simulator.

Minimal training impact as the value varied by only 4" in the simulator and operations were unaffected.

PERFORMANCE TEST ABSTRACT
TT-001

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-001, Manual Reactor Trip

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response following a manual reactor trip.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a manual Reactor trip is initiated. Various items of plant data are recorded for a period of time following the trip, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

6.1 SHNPP trip due to main generator fire, 10-9-89.

6.2 Panel of experts.

7.0 DATE PERFORMED/TEST RESULTS: 12-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT

TT-002

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-002, Simultaneous Loss of All Feedwater Pumps

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response following a loss of all main feedwater pumps.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a malfunction for a simultaneous loss of all feedwater pumps is initiated. Various items of plant data are recorded for a period of time following the trip, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 DATE PERFORMED/TEST RESULTS: 12-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-003

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-003, Simultaneous Closure of All Main Steam Isolation Valves

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response following a closure of all main steam isolation valves during power operation.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a simultaneous closure of all main steam isolation valves is activated. Various items of plant data are recorded for a period of time following the trip, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 DATE PERFORMED/TEST RESULTS: 12-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-004

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-004, Simultaneous Trip of All Reactor Coolant Pumps

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response following a trip of all reactor coolant pumps during power operation.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a simultaneous trip of all reactor coolant pumps is activated. Various items of plant data are recorded for a period of time following the trip, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 DATE PERFORMED/TEST RESULTS: 12-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-005

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-005, One Reactor Coolant Pump Trip

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

One, two, or three reactor coolant pumps may be tripped.

3.0 TESTED OPTIONS

Plant response following the trip of one reactor coolant pump during power operation.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a trip of one reactor coolant pump is activated. Various items of plant data are recorded for a period of time following the trip, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

6.1 Panel of experts.

6.2 SHNPP trip of "C" RCP at 100% power, 6-17-87.

7.0 DATE PERFORMED/TEST RESULTS: 12-13-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-006

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-006, Turbine Trip Below P-10

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response following the trip of the main turbine below the P-10 setpoint during power operation.

4.0 INITIAL CONDITIONS

Less than 10% power (P-10 setpoint), MOL.

5.0 TEST DESCRIPTION

After initialization a trip of the main turbine is activated. Various items of plant data are recorded for a period of time following the trip, to determine the capability of the simulator to perform correctly, and the test is then terminated. (P-10 setpoint is the power level below which a turbine trip will not cause a reactor trip.)

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 DATE PERFORMED/TEST RESULTS: 12-17-90/SAT

8.0 DEFICIENCES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-007

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-007, Maximum Rate Power Ramp

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response to the maximum power ramp rate (45 MW/minute) from 100% power to 75% and then back to 100%.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a maximum ramp rate is initiated from 100% to less than 75% power, and then a power increase, at the same ramp rate, back to 100% power. Various items of plant data are recorded to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 DATE PERFORMED/TEST RESULTS: 12-17-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-008

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-008, Maximum Size RCS Leak Inside Containment With Lost of Off-Site Power

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response to a maximum size RCS leak with a simultaneous loss of off-site power.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a malfunction to create the maximum size RCS leak, concurrent with a loss of off-site power is initiated. Various items of plant data are recorded for a period of time, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 **DATE PERFORMED/TEST RESULTS:** 12-14-90/UNSAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

SSR 91-0066, pressurizer refills with large break LOCA in progress.

The pressurizer level is very unstable, to the point that it refills. This is not physically possible and is a known problem with the RCS hydraulic model. The operators have been able to handle the situation. The RCS model is scheduled to be upgraded to resolve this problem by the end of 1992.

PERFORMANCE TEST ABSTRACT
TT-009

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-009, Maximum Size Steam Leak Inside Containment

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response to a maximum size stem leak inside containment.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a malfunction to create the maximum size steam leak, inside containment, is initiated. Various items of plant data are recorded for a period of time, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 **DATE PERFORMED/TEST RESULTS:** 12-14-90/SAT

8.0 DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT

None

PERFORMANCE TEST ABSTRACT
TT-010

1.0 PROCEDURE TITLE/ANSI 3.5 REFERENCE

1.1 TT-010, Slow RCS Depressurization to Saturation Using PORVs and No SI

1.2 ANSI/ANS-3.5, 1985, Section 4.2 and Appendix B, Section B.2.2

2.0 AVAILABLE OPTIONS

None

3.0 TESTED OPTIONS

Plant response to a slow RCS depressurization without safety injection.

4.0 INITIAL CONDITIONS

100% power, steady state, equilibrium xenon and decay heat, MOL.

5.0 TEST DESCRIPTION

After initialization, a malfunction to create a slow depressurization of the RCS to a saturated condition, without safety injection actuation, is initiated. Various items of plant data are recorded for a period of time, to determine the capability of the simulator to perform correctly, and the test is then terminated.

6.0 BASELINE DATA/REFERENCES

Panel of experts.

7.0 **DATE PERFORMED/TEST RESULTS:** 12-17-90/UNSAT

8.0 **DEFICIENCIES FOUND DURING TESTING AND TRAINING IMPACT**

SSR 91-0067, RCS flow step decrease at saturation.

When the RCS reaches saturation, RCS loop flows instantaneously drop to 50% of their initial values. This is recognized by the operators as a nonrealistic situation. This problem should be resolved by the end of 1992 with the scheduled RCS model upgrade.