NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

INITIAL

SIMULATOR CERTIFICATION SUBMITTAL

NOVEMBER 1990

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: Cooper Nuclear Station

DOCKET NUMBER: 50- 298

LICENSEE: Nebraska Public Power District

DATE: 11/30/90

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

Cooper Nuclear Station
Nuclear Training Center
Brownville, NE 68321

[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 attached Operability, Steady State, Transient and Malfunction Tests.

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

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

See attached test schedule

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

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

Not Applicable. Initial Submitted

[X] 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 Submitted

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

SIGNATURE – AUTHORIZED REPRESENTATIVE	TITLE	DATE
G. R. Horn [signature]	Nuclear Power Group Manager	11/30/90

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

Cooper Nuclear Station Simulator

EXCEPTIONS TO ANSI/ANS 3.5, 1985

(ANSI/ANS 3.5 - 3.2.1): VBD-FP, Sprinkler Control and Fire Alarm Panel was omitted. This panel is under review by the plant for Detailed Control Room Design Review enhancements. After the plant has determined which changes are to be made, then the simulator will evaluate for inclusion.

The seismic monitoring panel which has only limited operator interfaces will not be part of the simulated control room.

The control room ventilation radiation monitoring panel will not be part of the simulated control room. This panel has limited operator interfaces.

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

A.1 SIMULATOR INFORMATION

A.1.1 General

(1) The owner/operator of the simulator is Nebraska Public Power District. The simulator vendor was S-3 Technologies (previously Singer) of Columbia, Maryland.

(2) The reference plant is the Cooper Nuclear Station located near Brownville, Nebraska. Cooper Nuclear Station is a General Electric Boiling Water Reactor (BWR/4) rated at 2381 megawatts thermal.

(3) The simulator was available for training on June 4, 1990.

(4) This is Nebraska Public Power District's initial report for the CNS simulator.

A.1.2 Control Room

(1) The simulated control room replicates the plant control room to the greatest extent possible. The following exceptions are noted.

(a) The sound powered phone equipment mounted on VBD-FP, Sprinkler Control and Fire Alarm Panel, was relocated to the north wall of the simulator control room. This location is close proximity to the actual plant location.

(b) Wall mounted electrical/lighting panels, HVAC ducts, the kitchenette and toilet were omitted. There is no value to be gained by having these items in the simulator control room and this is considered a minor deviation.

(c) Diesel generator #2 local panels were added to the back panel row adjacent to Panel P-1. The panels installation configuration does not detract from training as they are not visible when viewing the control room panels from the front. The addition of these panels is considered to be a training enhancement.

A.1.2 Control Room - Continued

(d) The instructor station occupies the corner where the security access control room is located in the plant. The size of the instructor station necessitated a slight reduction in the size of the shift supervisor's office. Th' is considered a minor deviation with no impact on t.aining in the simulator.

(e) The local panels for safe shutdown of the reactor outside the control room are located in the instructor station booth. These local panels are not part of the plant control room but are considered training enhencements for the simulator control room.

(f) A Halon fire suppression system is used to protect the simulator computers and panels. The bottles sit in the simulated control room near the Diesel Generator panels out of view when operating the simulator control room panels. Smoke/fire detectors and Halon distribution nozzles are located in the simulator control room ceiling, unlike the pl' it.

(2) The simulated control panels duplicate the control room panels with the following exceptions.

(a) In order to accommodate obsolete or otherwise unavailable equipment, minor substitutions of equipment were allowed on a case by case basis. Each substitution request was reviewed for training and visual impact prior to approval. The substitution requests are on file and available for review.

(b) The Ronan annunciator alarm panels, CRTs and controls have been installed in the simulator. This installation was a known design change anticipated for future plant installation during simulator construction. The simulator still leads the plant in this area as the plant completed one of two segments for installing Ronan equipment during the 1990 Spring Refueling Outage. Completion of the Ronan Design Change for the control room equipment is scheduled for the 1991 Outage.

A.1.2 Control Room - Continued

(3) All systems are duplicated as configured at the time of simulator data freeze, May 5, 1988, except for Detailed Control Room Design Review (DCRDR) changes which had a data freeze of November 5, 1988. All logic, set point, and design changes have been tracked since data freeze. A Simulator Initial Upgrade Project is currently in progress. See section A.4.

(4) The simulated control room environment extensively duplicates the actual control room.

(a) Lighting levels reflect that of the actual control room.

(b) Lighting configuration responds to changes with the simulated power buses.

(c) Noises typically heard from the control room are duplicated.

(d) The instructor station windows are tinted to reduce its visibility by the control room crew.

A.1.3 Instructor Interface

The Cooper Nuclear Station simulator Acceptance Test Procedure (ATP) was developed to test and verify the simulator's capabilities as defined by ANSI/ANS 3.5 of 1985. The test commenced at the vendor's factory on September 20, 1989 and was completed on January 16, 1990. Site Reverification Testing was conducted between February 12 and February 24 and consisted of a rerun of selected portions of the ATP.

The Third Generation Instructor Station (TGIS) was used throughout the ATP to conduct the tests, store snapshots and Initial Conditions, override plant controls and indicators, and record transient data. The continuous use of TGIS during the ATP test confirmed the operability and reliability of the instructors interface.

A.1.3 Instructor Interface - Continued

(1) With the TGIS the instructor has at his/her immediate command 25 Initial Conditions (ICs) which were developed during Acceptance and Site Reverification Testing while performing the Mission Test (a plant startup and shutdown utilizing station procedures). Another 30 ICs are available for instructor storage of plant conditions for specific instruction or to carry events from one training session to the next. Of the ICs, 31 including the 25 mentioned above are password protected and cannot be changed by the instructors. A listing of the Initial Conditions is provided in Attachment A.

(2) The specified plant Malfunctions are listed in Attachment B. In addition to the malfunctions, the ability to provide failure of all simulated control room components is provided by the Input/Output (I/O) override via TGIS.

(3) Operations that are normally performed away from the control room in the plant are accomplished in the simulator by students/operators who establish normal communication with an instructor in the instructor's station. The instructor then initiates the expected response through the simulator's Remote Functions. These functions provide variable operation and appropriate time delays so that the operation appears to occur with the same response as in the plant's control room. A list of these functions is provided as Attachment C.

(4) TGIS features consist of the standard and special functions.

(a) Standard features and functions
Freeze - Run
Snapshot - Reset
Backtrack
IC Reset and Switch Check
Switch Check Override
Remote Function Control
Malfunction Control
Annunciator Control
Recorder Power Off
Daily Operational Readiness Test (DORT)
Simulator Startup/Shutdown
Computer Fault
Emergency Stop

A.1.3 Instructor Interface - Continued

(b) Special features and functions
Real Time Control - Slow/Real/Fast
Instructor Action Summary
I/O Override Control
Operator Action Monitor
Replay
Computer Assisted Exercise Program (CAEP)
Monitored Parameters
Video Screen Copy
Instrument Noise Disable
Roll - Around TGIS Console
Dynamic Representation of all Control Boards and Panels

The simulator provides a method of alerting the instructor that the simulator has exceeded normal simulator operating limits (i.e., conditions that are unanalyzed). If one of the simulator operating limits is exceeded, the simulator is placed in a freeze condition and a "pop up" message is displayed on the Third Generation Instructor Station that indicates which operating limit has been exceeded. The operating limits are:

Primary Containment Pressure
Suppression Pool Water Temperature
Fuel Temperature
Reactor Pressure Vessel Pressure

A.1.4 Operating Procedures

Cooper Nuclear Station plant procedures were used to conduct the Acceptance Test Procedure (ATP). The ATP outlined the activities, the expected response and provided sign off space for the test after the simulator met the critical objective. The plant procedures were used to perform the tasks of the ATP mission. The plant procedures were also used to specify operator response to transients and malfunctions. The only significant difference in procedures resulted from the simulator leading the plant with the installation of the new Ronan annunciator alarm system. As a result, alarm locations and actual engravings may vary from those defined in the plant procedures. A cross reference was developed to overcome this difficulty in the simulator until the plant has completed the upgrade to the Ronan Annunciators during the 1991 Refueling Outage. Having the new Ronan annunciator system in place will give the operators an advantage or "heads up" in using the new system.

A.1.5 Changes Since Last Report

As this is the initial report there have been no changes to the simulator.

A.2 SIMULATOR DESIGN DATA

The Cooper Nuclear Station Simulator Design Data Base index list is a two-inch thick computer print-out. This is available for review on request.

A.3 SIMULATOR TESTS

A.3.1 Computer Real Time Tests

The computer incorporates a utility program to monitor real time simulation, TIMMEM. TIMMEM is a monitoring program used to obtain time usage statistics on all individual program modules on the simulation computer. This program was used during Factory Acceptance Testing to verify that the simulator operates in real time and that sufficient time and memory exists to support expansion without exceeding real time simulation.

The test began with a full power end-of-core life Initial Condition. A scram signal was introduced and appropriate operator action was taken to stabilize the plant. The test was conducted for fifteen minutes and a print out of each modules memory utilization was generated. The IPU executive programs were then moved to the CPU and the test was performed again and another print out obtained. The test results are documented in the Acceptance Test Procedure, and shows sufficient spare execution time.

The test results provided a measurement of spare main memory as well as the time utilized by each module.

A.3.2 Steady State and Normal Operations Test

A one hour Stability Test at 100% power and normal operation tests at 30%, 53%, and 75% power were conducted as part of the Acceptance Test Procedure. All tests were satisfactorily completed. Individual, specific results are available at the Cooper Nuclear Station Training Center upon request. The following critical parameter data was recorded during the test:

Reactor Thermal Power	Torus Water Temperature
Neutron Flux	Primary Containment Pressure
Reactor Pressure	Primary Containment Temperature
Reactor Water Level	Main Steam Flow
Recirculation Loop Flow	Main Steam Pressure
Recirculation Loop Temperature	Feedwater Flow
Total Core Flow	Feedwater Temperature to Reactor
Total Core Differential Pressure	Main Condenser Vacuum
Torus Level	Generator Megawatts

A.3.3 Transient Tests

The transients tested duplicated those outlined in Appendix B.1 of ANSI/ANS 3.5 of 1985. During the tests, the parameters listed in ANSI/ANS 3.5, Appendix B.1.2 were recorded. Data from the simulator tests was compared against plant data from the station's Plant Management Information System computer archive files of similar plant events for the following tests: 1, 2, 4, 5, and 6.

The tests were:

1) Manual scram
2) Simultaneous trip of all feedwater pumps
3) Simultaneous closure of all MSIVs
4) Simultaneous trip of both recirculation pumps
5) Single recirculation pump trip
6) Turbine trip
7) Maximum rate power ramp (decrease followed by increase)
8) Design basis LOCA with Loss of Offsite Power
9) Unisolable steamline break
10) Composite scenario (See Note 1)

As referenced in the Design Database Report, a combination of startup test data, plant reference archive data, and operational assessment of USAR data were used in these tests.

A.3.4 Malfunction Tests

All malfunctions were tested during the Factory Acceptance Test utilizing plant procedures as the basis for operator action. The Malfunction Cause and Effect Manual was developed using the plant procedures, Updated Safety Analysis Report, operation experience and input, and engineering evaluations to predict the plant response.

A list of the specific malfunctions is included as Attachment B.

A.4 SIMULATOR DISCREPANCIES AND PLANT UPGRADE

Since the simulator data freeze, Simulator Design Change Requests (SDCRs) have been written documenting all changes to the reference plant. Most were written as a result of reviews of plant change packages or Discrepancy Reports from simulator operation. These SDCRs were evaluated and prioritized for the current Simulator Initial Upgrade Project that will configure the simulator up to date through the 1990 plant outage. Bids have been received for the Initial Upgrade Project, and it is scheduled to begin in the last quarter of 1990.

A complete design data base is maintained for the simulator design control. This data base contains all drawings, technical manuals, recorder strip charts, accident analysis and design calculations used to design and build the simulator. This data base provides the basis from which the simulator is modified to maintain it current with the reference plant.

To ensure that simulator design data is maintained current with respect to the plant, controlled procedures and controlled drawings are maintained.

Reactor scram records, plant licensee event reports (LERS), INPO SERs and SOERs and NRC bulletins/IE Notices are reviewed for simulator configuration and or performance impact.

Student feedback is monitored to implement training course critique comments into simulator configuration and/or performance.

A simulator computerized configuration management system (CMS) is used to track all plant modifications. This system logs and maintains current status of each modification.

A.5 SUMMARY

In summary, this report when used in conjunction with the simulator documentation and referenced documentation such as Simulator Design Specifications, Acceptance Test Procedures, performance data, etc., provides conclusive evidence that:

1. The design of the simulator was in conformance with the design of the reference plant as of the Design Data Freeze date of May 5, 1988. The design fully meets or exceeds the intent of ANSI/ANS 3.5. This report also discusses NPPD's intent and plans to incorporate plant changes past data freeze into the simulator with the intent to fully meet or exceed the standards of ANSI/ANS 3.5.

2. Formal test programs were employed throughout the construction of the simulator to ensure that the design of the simulator fully met or exceeded the performance related requirements of ANSI/ANS 3.5.

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 9/20/89 - 9/?5/89

Description of Test: Plant Startup - cold to hot standby.

Reference to Standard: ANSI/ANS 3.5 Section 3.1.1(1), Section 3.1.1(5) and Section 3.1.1(7)

Available Options: The initial condition accounts for the preparation of the startup procedure. The listed procedures/sections are reflected in the control panel indications:

SOP 2.1.1 Att-A
SOP 2.1.1.2

Tested Options: Cold shutdown conditions up to the point where the mode switch is placed in Run.

Initial Conditions: IC01 - Cold shutdown configuration at Beginning of Life (BOL), moderator temperature 110 F, no decay heat. All rods are fully inserted. Core is xenon-free. Shutdown cooling is operating. RR pumps are shutdown.

Final Conditions: IC09 - Plant is at 650 psig with reactor power corresponding to a heat up rate of 60 F/hr. Xenon following core physics. One feed pump providing feed water flow in automatic control. Bypass valves controlling pressure. SJAEs operating.

Description of Data to Determine Fidelity: Plant experience of the operators conducting the test. Utilized reference plant procedures:

S.P 6.1.24	SOP 2.2.56	SOP 2.2.77
SOP 2.2.1	SOP 2.2.74	SOP 2.2.49
SOP 2.2.9	SOP 2.2.8	SOP 2.2.66
SOP 2.2.69	SOP 2.2.70	SOP 2.2.40
SOP 2.2.68	SOP 2.2.85	SOP 2.2.68.1
SOP 2.2.69B	NPP 10.13	SOP 2.2.8
SOP 2.2.77	SOP 2.2.52	SOP 2.2.51
SOP 2.2.3	SOP 2.2.6	SOP 2.2.5
SOP 2.2.75	SOP 2.2.55	SOP 2.2.67
SOP 2.2.33	SOP 2.2.80	SOP 2.2.77
SOP 2.2.28	SOP 2.2.25	

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 9/26/89 - 9/28/89

Description of Test: Nuclear startup from hot standby to rated power

Reference to Standard: ANSI/ANS 3.5 Section 3.1.1(2), Section 3.1.1(3), Section 3.1.1(6), and Section 3.1.1(7)

Available Options: Performance of Station Operating Procedures utilized for normal startup of the plant from hot standby to rated power.

Tested Options: Normal startup of the plant from hot standby to rated power.

Initial Conditions Plant is at 650 psig with reactor power corresponding to a heatup rate of 60 F/hr. Xenon following core physics. One feedpump providing feedwater flow in automatic control. Bypass valves controlling pressure. SJAEs operating.

Final Conditions: Steady state full power condition, xenon at equilibrium.

Description of Data to Determine Fidelity: Plant experience of the operators conducting the test. Utilized reference plant procedures:

SOP 2.1.1	SOP 2.2.60.1	SOP 2.2.60
SOP 2.2.47	SOP 2.2.77	SOP 2.2.53
SOP 2.2.14	SOP 2.2.28	SOP 2.2.16

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 9/29/89

Description of Test: Plant shutdown from rated power to hot standby and cooldown to cold shutdown conditions.

Reference to Standard: ANSI/ANS 3.5 Section 3.1.1(8), Section 3.1.1(7)

Available Options: Beginning of Life (BOL), Middle of Life (MOL), and End of Life (EOL) core exposures.

Tested Options: Test utilized EOL core exposure.

Initial Conditions: Plant at steady state power full power condition, xenon at equilibrium.

Final Conditions: Plant in shutdown cooling mode, reactor pressure vessel vented.

Description of Data to Determine Fidelity: Plant experience of the operators conducting the test. Utilized reference plant procedures:

SOP 2.1.10	SOP 2.1.4	SOP 2.2.6
SOP 2.2.28	SOP 2.2.77	SOP 2.2.13
SOP 2.2.69	SOP 2.2.69.2	SOP 2.2.25
SOP 2.2.56	SOP 2.2.3	SOP 2.2.75

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 10/03/89 - 10/05/89

Description of Test: Core Performance Testing

Reference to Standard: ANSI/ANS 3.5 Section 3.1.1(9)

Available Options: Testing divided as follows: Shutdown Margin Test, Reactivity Feedback Test, Rod Density Verification and Flux profiles, Kinetics Response, and Region Relative Power.

Tested Options: Tested all options

Initial Conditions: Moderator temperature 178 degree F, heating up from RR pumps. Reactor is approaching criticality at approximately 2 rods subcritical. Xenon is following core physics. Shutdown cooling is secured, both RR pumps running at minimum speed.

Final Conditions: Not applicable for core performance testing.

Description of Data to Determine Fidelity: Operators were assisted by engineering and utilized plant reference data:

GE 23A4782 NEDE-31153

BOC-11 Tip Traces (DR25-S)

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 10/03/89 - 12/20/89

Description of Test: Operator Conducted Surveillance Testing

Reference to Standard: ANSI/ANS 3.5 Section 3.1.1(10)

Available Options: The below listed surveillances can be run any time during core life.

S.P. 6.1.2	S.P. 6.1.3	S.P. 6.1.6
S.P. 6.1.7	S.P 6.3.2.1	S.P. 6.3.4.3
S.P. 6.3.5.1	S.P. 6.3.6.1	S.P. 6.4.8.2.4

Tested Options: Surveillance were conducted at various core life.

Initial Conditions: Initial Condition dependant upon power requirements to meet the conditions for conducting the surveillance.

Final Conditions: Not applicable for testing of the surveillances.

Description of Data to Determine Fidelity: Plant experience of the operators conducting the surveillances. Utilized plant reference procedures that are not otherwise referenced or tested elsewhere during the Acceptance Test Procedure. Surveillances selected involve multiple system interfaces, and are best observed on an integrated plant.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 1/05/90

Description of Test: Steady State Performance (30%, 53%, 75%,100%)

Reference to Standard: ANSI/ANS 3.5 Section 4.1 and Appendix B Section B1.1(1) and (2)

Available Options: Middle of Life (MOL) and End of Life (EOL). Power levels for which plant reference heat balance data is available.

Tested Options: 30% and 100% conducted at MOL. 53% and 75% conducted EOL.

Initial Conditions: Initial Condition for 30% Power: RR pumps at minimum flow. Xenon building in. Plant operating point is just below the instability region. Main generator is tied to the grid. Drywell being inerted.
IC for 53% Power: Power is just to the right of the instability region at 100% rod pattern. Xenon is at equilibrium. Drywell inerting is complete.
IC for 75% Power: Same as for 53 % power except that power is raised to desired level by increasing recirculation pump flow.
IC for 100 % Power: Steady state full power operation, xenon at equilibrium.

Final Conditions: The simulator computed values for steady state, full power operation with the reference plant control system configuration shall be stable and not vary more than +/- 2% of the initial values over a 60 minute period. The simulator computed values of critical parameters shall agree with +/- 2 of the reference plant parameters and shall not detract from training. The accuracy of computed values shall be determined for the three points over the power range.

Description of Data to Determine Fidelity: Data used for determination of fidelity was Plant Management Information System data, plant analysis data (Updated Safety Analysis Report) and plant design data.

Deficiencies Noted During Test: Average Condenser Total Pressure had a deviation of 8.37%. Simulator computed a value of 1.37 psia and plant design data is 1.50 psia. This delta is negligible and does not detract from training.
Average Temperature of Drywell had a deviation of 5.36%. Simulator computed a value of 142.18 degree F as derived from actual plant drywell temperature profiles and this value was compared to plant analysis data of 135 degree F. This delta is insignificant and does not detract from training.

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/1/89

Description of Test: Manual Scram From 100% Power

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(1)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, beginning of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Transient is stopped at the point where the high pressure injection systems of HPCI, RCIC, and feedpumps trip on high water level.

Description of Data to Determine Fidelity: Transient was compared to similar plant data where the plant was manually scrammed at 35% power. This plant data was captured via the Plant Management Information System.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/4/89

Description of Test: Simultaneous Trip of All Feedwater Pumps

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(2)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Transient is stopped at the point where high pressure injection systems, HPCI and RCIC, feed the vessel up to the high water level trip of these systems.

Description of Data to Determine Fidelity: Transient was compared to a plant scram due to loss of feedwater. Plant was at approximately 80% power prior to the scram.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/4/89

Description of Test: Simultaneous Closure of All MSIV's

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(3)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: All MSIVs are closed, pressure being controlled by low-low set, reactor water level increasing due to injection by HPCI and RCIC.

Description of Data to Determine Fidelity: Simulator traces compared to USAR main steam line break accident analysis for general trends.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/19/89

Description of Test: Simultaneous Trip of All Recirculation Pumps

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(4)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% Full Power, Middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Reactor power decrease to approximately 43.5%; no APRM high flux rod block. Reactor pressure decrease to approximately 945-950 psig. Reactor vessel level control system responds to level increase and decreasing steam flow by reducing total feedwater flow. Plant stabilizes and no plant trip.

Description of Data to Determine Fidelity: Compared key parameters with CNS Restart Testing Final Summary Report NEDC-31304.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

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PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/19/89

Description of Test: Single Recirculation Pump Trip

Reference to Standard: ANSI/ANS 3.5 Appendix B Section B1.2(5)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Reactor power decrease to approximately 65% power; no APRM high flux rod block. Reactor pressure decrease to approximately 970 psig. Reactor vessel level control system responds to level increase and decreasing steam flow by reducing total feedwater flow. Plant parameters are stabilizing with no operator action within about one minute after transient initiation.

Description of Data to Determine Fidelity: Simulator traces compared to CNS Restart Testing Final Summary Report NEDC-31305 and plant data of a trip of a single recirculation pump.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/19/89

Description of Test: Single Recirculation Pump Trip

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(5)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Reactor power decrease to approximately 65% power; no APRM high flux rod block. Reactor pressure decrease to approximately 970 psig. Reactor vessel level control system responds to level increase and decreasing steam flow by reducing total feedwater flow. Plant parameters are stabilizing with no operator action within about one minute after transient initiation.

Description of Data to Determine Fidelity: Simulator traces compared to CNS Restart Testing Final Summary Report NEDC-31305 and plant data of a trip of a single recirculation pump.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/20/89

Description of Test: Main Turbine Trip

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(6)

Available Options: An Initial Condition where the transient is not terminated due to a direct reactor scram from the turbine trip.

Tested Options: Power at 35% being increased by rod withdrawal. RR pumps at minimum flow. Xenon building in. Plant operating point is just below the instability region. Main generator is tied to the grid. Drywell being inerted.

Initial Conditions: Simulator is set up in the configuration stated in the Tested Options. Reduce power with control rod insertion until the direct scram from a main turbine trip is bypassed. TCV Fast Close and TSC Trip Bypass annunciator actuated.

Final Conditions: Feedwater system temperatures begin to decrease due to loss of extraction steam. As feedwater temperature to the reactor is decreasing, reactor power and pressure begin to increase slowly. Main turbine bypass valves open until fully open. Steam flow and feed flow increase. Feed flow increase is limited by Startup Flow Control valve capacity. Reactor level begins to decrease. Once the main turbine bypass valve are full open, reactor pressure increase is more rapid, reactor power increase is more rapid. Reactor scram occurs from either high reactor pressure or high APRM flux.

Description of Data to Determine Fidelity: Simulator traces and response is compared against USAR for general trends. Transient was reran from 75% power to compare against plant data from a turbine trip and scram event of 5/18/87.
Results were satisfactory as compared to the reference plant data.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/19/89

Description of Test: Maximum Rate Power Ramp (master recirculation flow controller in "manual") Down To Approximately 75% And Back Up To 100%

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(7)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Plant stabilizes near initial condition after the power ramp. No scram occurs.

Description of Data to Determine Fidelity: Simulator traces and response was compared to plant data for a reactor recirculation pump runaway event at reference plant. Verified that parameter response is similar to reference plant data.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/6/89

Description of Test: Maximum Size Reactor Coolant System Rupture with Loss of All Offsite Power.

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(8)

Available Options: Pipe break can be of either reactor recirculation discharge loops. Transient is conducted at 100% full power. Selection can be made from core life of Beginning, Middle, or End.

Tested Options: 100% full power, middle of core life. Reactor recirculation pump A discharge line rupture.

Initial Conditions: The following malfunctions are activated simultaneously. RR32-A (RR A discharge loop rupture), ED05 (Loss of Startup Transformer), ED06 (Loss of Emergency Transformer), ED07 (Loss of Normal Transformer), ED15 (Loss of 12.5 KV Power).

Final Conditions: Core spray injection begins at approximately 340 psig. LPCI injection begins at approximately 268 psig. Level is restored and maintained at or above the top of the fuel by low pressure CSCS systems.

Description of Data to Determine Fidelity: Simulator traces and response is compared against similar event in USAR, verifying for general trends.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/6/89

Description of Test: Steam Line Break (Unisolable)

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(9)

Available Options: Any one of the four main steam lines upstream of the inboard MSIV can be selected for the rupture from 100% full power. Core life can be selected from Beginning, Middle, or End.

Tested Options: 100% full power, middle of core life. Main steam line "A" was selected for the rupture.

Initial Conditions: 100% full power, malfunction MS02-A (Steam Line 'A' Rupture Inside Primary Containment) is activated at 100% severity.

Final Conditions: Low pressure CSCS pumps sequentially start. Low pressure injection increases reactor level until interrupted by operator action.

Description of Data to Determine Fidelity: Simulator traces and response were compared to a similar event in the USAR, verifying for general shape of the trends in simulator compared to the USAR.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

COOPER NUCLEAR STATION SIMULATOR
ANSI/ANS 3.5 CERTIFICATION REPORT
INITIAL REPORT, NOVEMBER 1990

PERFORMANCE TEST ABSTRACT

Date Test Conducted: 12/6/89

Description of Test: Composite Scenario - Transient consist of a simultaneous closure of all MSIVs combined with a single stuck open safety/relief valve and high CSCS injection systems (HPCI and RCIC) disabled.

Reference to Standard: ANSI/ANS 3.5 Appendix B, Section B1.2(10)

Available Options: 100% full power Initial Conditions for core life of Beginning, Middle and End.

Tested Options: 100% full power, middle of core life.

Initial Conditions: Steady state full power operation, xenon at equilibrium.

Final Conditions: Reactor pressure decrease to approximately 200 psig. Wide range water level decrease to approximately 90 inches TAF. LPCI and CS do not inject.

Description of Data to Determine Fidelity: Simulator traces and response were compared to similar events in the USAR. Although there are not unique events to compare to, the MSIV closure was the closest to this event. The stuck open relief valve allowed the vessel mass to decrease and depressurize, which was the expected response.

Deficiencies Noted During Test: None

Exception(s) Taken to ANSI/ANS 3.5 (With Justification): None

TEST SCHEDULE

MFSCERT1

MALFUNCTION TESTING SCHEDULE - DEC '90 THROUGH NOV '91

DEC '90	JAN '91	FEB '91	MAR '91	APR '91	MAY '91	JUN '91	JUL '91	AUG '91	SEP '91	OCT '91	NOV '91
CR01	DG01	ED08A	ED15	FW05	IA01	NM03	RD02	RH01	RR04	SW01	SW04
CS01	DG02	ED08B	FW01	HP01	MC01	NM04	RD06	RP01	RR05		TC01
	ED03	ED08C			MS01			RP02			
	ED12A	ED08D			MS02						
	ED12B	ED08E			NM01						
	ED12C	ED08F			NM02						
	ED12D	ED08G									

MALFUNCTION TESTING SCHEDULE - DEC '91 THROUGH NOV '92 MFSCERT2

DEC '91	JAN '92	FEB '92	MAR '92	APR '92	MAY '92	JUN '92	JUL '92	AUG '92	SEP '92	OCT '92	NOV '92
CS02	DG03	ED14A	FW09	FW13	MS03	PC04	RD10	RH04	RP05	RR16	
	ED04	ED14B	FW11	HP04	NM05	RD09	RD12	RP04		SW05	
	ED09A	ED13A			NM06		RD13			SW07	
	ED09B	ED13B			NM07						
	ED09C	ED13C			NM08						
	ED09D	ED13D									
	ED09E	ED16									

MALFUNCTION TESTING SCHEDULE - DEC '92 THROUGH NOV '93 MFSCERT3

DEC '92	JAN '93	FEB '93	MAR '93	APR '93	MAY '93	JUN '93	JUL '93	AUG '93	SEP '93	OCT '93	NOV '93
CR03	ED05	ED10A	ED17	FW18A	MS08	NM12	RD20	RP06	RR17	RR33A	SW11
	ED06	ED10B	FW16	FW18B	NM09			RP07	RR20	SW09	TC07
	ED12E	ED10C		FW19						SW10	TC09
	ED12F	ED10D		FW20							
	ED13E	ED10E		IA03							
		ED10F									
		ED10G									
		ED10H									

MALFUNCTION TESTING SCHEDULE - DEC '93 THROUGH NOV '94 MFSCERT4

DEC '93	JAN '94	FEB '94	MAR '94	APR '94	MAY '94	JUN '94	JUL '94	AUG '94	SEP '94	OCT '94	NOV '94
AD06	ED07	ED11F	EG09		MS10		RD26	RR25	RR27	SW15	
CU08	ED11A	ED11G	FW21		NM13		RD27	RR26	RR28	SW16	
CU09	ED11B	ED11H	FW22		NM14		RH07		RR29	TC12	
	ED11C	ED11I	FW23		NM15				RR30		
	ED11D	ED11J							RR31		
	ED11E	ED13F							RR32		
									RR33B		

OPERABILITY TESTING SCHEDULE (ANNUAL) OPTESTSCHED

NOTE: SS = STEADY STATE
STAB = STABILITY
TRANS = TRANSIENT

DEC '90	JAN '91	FEB '91	MAR '91	APR '91	MAY '91	JUN '91	JUL '91	AUG '91	SEP '91	OCT '91	NOV '91
TRANS #1				TRANS #5				TRANS #9			
						MISSION TEST (3.1.1)					
	TRANS #2				TRANS #6				TRANS #10		
		TRANS #3				TRANS #7				STAB 100%	
SS 25%				SS 50%				SS 100%			
			TRANS #4				TRANS #8			REAL TIME TEST	

Attachment A

SIMULATOR INITIAL CONDITIONS

The following Initial Conditions (IC's), Rev. 3, were developed during the week of February 12, 1990. The required conditions were established using the following references:

1. NPPD Simulator Spec, Appendix C
2. SIM-88-253
3. SIM-89-012
4. Site Reverification discussions

Each IC lists the IC title, critical parameter values, and a brief narrative of the IC.

IC01 Cold Shutdown (BOL)
IC02 Cold Shutdown (MOL)
IC03 Cold Shutdown (EOL)
IC04 Cold Shutdown (BOL)
IC05 Cold Shutdown (MOL)

IC06 Cold Startup (EOL)
IC07 Critical Heatup (BOL)
IC08 Feedpump Startup (BOL)
IC09 Transfer to Run (BOL)
IC10 Main Turbine Startup (BOL)

IC11 Generator Synchronization (BOL)
IC12 Start 2nd Feedpump (BOL)
IC13 Power Increase from 60% (BOL)
IC14 60% Steady State (BOL)
IC15 Full Power Operation (BOL)

IC16 Full Power, Xenon Buildup (BOL)
IC17 Full Power Operation (MOL)
IC18 Full Power Operation (EOL)
IC19 Power Decrease (BOL)
IC20 Continue Power Decrease (BOL)

IC21 Shutdown Cooling Preparation (BOL)
IC22 Hot Standby (BOL)
IC23 Hot Scram Recovery (BOL)
IC24 Cold Scram Recovery (BOL)
IC25 Instability Region (BOL)

Attachment B

SIMULATOR MALFUNCTIONS

AD01 ADS Timer Failure

AD02 ADS Timer Inadvertent Start

AD03 Low Low Set Fails to Initiate

AD04 SRV Tailpiece Pressure Switch Failure

AD05 SRV Tailpiece Vacuum Breaker Failure

AD06 Reactor Pressure Relief Valve Complete Failure

AD07 Reactor Pressure Relied Valve Partial Failure

CR01 Fuel Cladding Failure

CR02 Increased Control Rod Worth

CR03 Gross Fuel Cladding Failure

CR04 Low-flow High-power Instability (LaSalle)

CS01 Core Spray Pump Trip

CS02 Core Spray Injection Valve Fails to Auto Open

CS03 Core Spray Pump Discharge Line Break (Inside Vessel)

CS04 Core Spray Pump Discharge Line Break (Outside Vessel)

CU01 RWCU Pump Seal Failure

CU02 RWCU Non-regenerative Heat Exchanger Tube Leak

CU03 RWCU Blowdown Flow Control Valve Failure

CU04 RWCU Pump Trip

CU05 RWCU Filter/Demin Resin Intrusion into Reactor Vessel

CU06 RWCU Filter/Demin Resin Depletion

CU07 RWCU Filter/Demin Resin High DP

CU08 RWCU System Leak Inside Containment

CU09 Cold Water Cleanup Leak

DG01 Diesel Generator Fails to Start

DG02 Diesel Generator Trip

DG03 Diesel Generator Breaker Fails to Close (Auto)

DG04 Diesel Generator Voltage Regulator Failure

DG05 Diesel Generator Governor Failure

ED01 and ED02 have been deleted

ED03 4160V Bus Auto Transfer Failure

ED04 Loss of All AC Power

ED05 Loss of Power (Start-up Transformer)

ED06 Loss of Power (Emergency Transformer)

ED07 Loss of Power (ED03 L Transformer)

ED08 480V Power Bus Failure

ED09 480V Power Bus Failure

ED10 MCC Failure

ED11 Critical 120 VAC Power Panel Failure

ED12 Loss of 125 VDC Bus

ED13 Loss of 250 VDC Bus

ED14 Loss of 24 VDC

ED15 Loss of 12.5 KV Power

ED16 Battery Charger Failure (125 VDC)

ED17 Loss of Non-break Power Panel No.1

ED18 PMIS System Power Failure

EG01 Main Generator Field Breaker Fails to Open

EG02 Main Transformer Loss of Cooling

EG03 Generator Hydrogen Oil System Failure

EG04 Hydrogen Seal Oil System Failure

EG05 Main Generator Auto Voltag. Regulator Signal Failure

EG06 Loss of Generator Exciter Cooling

EG07 Loss of Generator Hydrogen Cooling

EG08 Total Load Loss

EG09 Main Generator Trip

FW01 Reactor Feedwater Pump Trip

FW02 Reactor Feedwater Pump Loss of Lube Oil Pressure

FW03 Reactor Feedwater PUmp Minimum Flow Valve Failure

FW04 Exhaust Hood Spray Valve Failure

FW05 Reactor Feedwater Pump Individual Controller Failure

FW06 Reactor Feedwater Pump Individual Controller Oscillation

FW07 Reactor Feedwater Pump Vibration

FW08 CSCS Pressure Maintenance Failure

FW09 Reactor Feedwater Pump Master Controller Oscillation

FW11 Main Feedwater Flow Signal Failure to RVLC System

FW12 Main Feedwater Flow Signal Oscillation

FW13 Steam Flow Signal Failure to RVLC System

FW14 Condensate Pump Trip

FW15 Condensate Booster Pump T.ip

FW16 Hotwell Level Controller (Auto) Failure

FW17 Condensate PUmp suction Boot Rupture

FW18 Feedwater Line Break (Inside Containment)

FW19 Feedwater Line Break (Outside Containment)

FW20 High Pressure Feedwater Heater Level Control Failure

FW21 Feedwater Control Signal Failure

FW22 Main Feedwater Startup Master Controller Failure

FW23 Lovejoy Feedwater Controller Failure

FW24 Lovejoy Feedwater Controller Oscillation

FW25 High Pressure Feedwater Heater Tube Leak

FW26 3-4 Feedwater Heater Tube Leak

FW27 1-2 Feedwater Heater Tube Leak

HP01 HPCI System Failure to Auto Start

HP02 HPCI Turbine Trip

HP03 HPCI Discharge Line Break

HP04 HPCI Flow Controller Failure

HP05 HPCI Inadvertent Initiation

HP06 HPCI Steam Line Break

HP07 HPCI System Discharge Header Back Leakage

HP08 HPCI Suction Transfer Failure

HV01 Drywell Fan Coil Unit Trip

IA01 Instrument Air System Depressurization

IA02 Service Air System Depressurization

IA03 Non-critical Instrument Air System Leak

IA04 Air Compressor Trip

IA05 IA Filter/Dryer Clogging

MC01 Main Condenser Air In Leakage

MC02 Main Condenser Tube Leak

MC03 Main Condenser Tube Flow Blockage

MC04 Steam Jet Air Ejector Steam Supply Pressure Control Fail

MC05 Circulation Water Pump Trip

MC06 Traveling Screen Blockage

MC07 SJAE Condenser High Level

MC08 Circ Water Leakage in Turbine Building Basement

MS01 Steam Leakage Inside Primary Containment

MS02 Steam Line Rupture Inside Primary Containment

MS03 Steam Line Rupture Outside Primary Containment

MS04 Gland Seal Steam Exhauster Condenser High Level

MS05 Steam Seal Pressure Regulator Failure

MS06 MSIV Disc Separates from Steam

MS07 MSIV Failure (MSIV Fails As Is)

MS08 DEH System Pressure Transmitter Failure

MS09 MSIV Closure Time Malfunction

MS10 Steam Leakage in Turbine Building

NM01 Rod Block Monitor Failure

NM02 SRM Channel Failure

NM03 SRM Channel Retract Permit Failure

NM04 SRM Channel Detector Stuck

NM05 IRM Channel Failure

NM06 IRM CHannel Detector Stuck

NM07 IRM Channel Retract Permit Failure

NM08 IRM Failure

NM09 APRM Signal Failure

NM10 TIP Probe Detector Stuck In Core

NM11 TIP Dry Tube Leak

NM12 SRM Inop

NM13 IRM Inop

NM14 APRM Inop

NM15 APRM Flow converter Unit Inop Failure

OG01 Explosion (Fire) in Air Ejector Discharge Piping

OG02 Sustained Hydrogen Burn

PC01 Nitrogen System Supply Failure

PC02 Torus to Drywell Vacuum Breaker Failure

PC03 Suppression Pool Water Temperature Instrument Failure

PC04 Instrument Line Break

PC05 Torus Air Temperature Instrument Failure

PC06 SGT System Flow Restriction

PC07 Leak from Torus Air Space

PC08 Suppression Pool Water Leak

PC09 Leak in Primary Containment

PC10 High Drywell Pressure Scram Signal

PC11 High Drywell Pressure CSCS Initiation Signal

RC01 RCIC System Failure to Auto Start

RC02 RCIC Turbine Trip

RC03 RCIC Discharge Line Break

RC04 RCIC Flow Controller Failure

RC05 RCIC Inadvertent Initiation

RC06 RCIC Steam Line Break

RD01 SDV Drain/Vent Valve Failure

RD02 ATWS

RD03 Scram Outlet Valve Leakage

RD04 CRD System Flow Control Valve Failure

RD06 CRD Drive Water Filter Clogging

RD07 CRD Stabilizing Valve Failure

RD08 CRD Hydraulic Pump Trip

RD09 Control Rod Drift In

RD10 Control Rod Drift Out

RD11 Control Rod Accumulator Trouble

RD12 Control Rod Stuck

RD13 Control Rod Uncoupled

RD14 Control Rod Scrammed

RD15 Control Rod Failure to Scram

RD16 Control Rod Slow Scram Time

RD17 Control Rod RPIS Reed Switch Failure

RD18 Loss of Air to Control Rod Drive HCUs

RD19 Rod Drive Control System Timer Malfunction

RD20 RPIS Failure (Total)

RD21 RPIS Failure (Partial)

RD22 Scram Discharge Volume Rupture

RD23 Abnormal Rod Motion Speed; Fast

RD24 Abnormal Rod Motion Speed; Slow

RD25 CRD Withdraw Header Vent Line Failure

RD26 Backup Scram Valve Failure

RD27 Automatic ARI Failure

RH01 RHR Pump Trip

RH02 RHR System Discharge Header Back Leakage

RH03 RHR Heat Exchanger tube Leak

RH04 LPCI Inboard Injection Valve Failed Closed

RH05 RHR System Discharge Line Break

RH06 LPCI Outboard Injection Valve Failed Open

RH07 RHR Shutdown Cooling Valve Isolation

RM01 Process Liquid Radiation Monitor Failure

RM02 Gaseous Radiation Monitor Failure

RM03 Area Radiation Monitor Failure

RP01 Scram Pilot Solenoids Fail to Deenergize

RP02 Reactor Scram

RP03 Electrical Protection Assembly Trip

RP04 Group - 1 Isolation Failure

RP05 Group - 2 Isolation Failure

RP06 Group - 3 Isolation Failure

RP07 Group - 6 Isolation Failure

RP08 Spurious PCIS Group - 1 Isolation Signal

RP09 Spurious PCIS Group - 2 & 6 Isolation Signal

RP10 Spurious PCIS Group - 3 Isolation

RP11 Scram Pilot Solenoids Deenergize

RR01 RR Pump SEal #1 Restrictive Orifice Plugging

RR02 RR PUmp Seal #2 Restrictive Orifice Plugging

RR03 Recirc Pump Shaft Binding

RR04 Recirc Pump Drive Motor Breaker Trip

RR05 Recirc Pump Field Breaker Trip

RR06 Recirc Pump Control Signal Failure

RR07 REcirc Pump Incomplete Start Sequence

RR08 RRMG Scoop Tube Lockout Failure

RR09 Recirc Motor Generator Tachometer Failure

RR10 Recirc Pump #1 Seal Failure

RR11 Recirc Pump #2 Seal Failure

RR12 Recirc Pump Runback (20%)

RR13 Recirc Pump Runback (45%)

RR14 Recirc Loop Flow Transmitter Failure

RR15 Jet Pump Flow Nozzle Blockage

RR16 Recirc Master Flow Controller Failure

RR17 Recirc M/A Flow Controller Failure

RR18 Jet Pump Failure

RR19 Shutdown Cooling Loss of Mass

RR20 Coolant Leakage Inside Primary Containment

RR21 Vessel Head Inner Seal Leakage

RR22 RRMG Set 1A Oil Pump Trip

RR23 RRMG Set 1B Oil Pump Trip

RR24 RPT Fails to Initiate Recirc Pump Trip

RR25 LIS-57/LIS-58)NR Yarway) Fails Downscale

RR26 LIS-72/A/B/C/D (NR Yarway) Fails Downscale

RR27 LT-52 (NR GEMAC) Failure

RR28 LIS-101 (NR Barton) Failure

RR29 LIS-83 Switch Fails Upscale

RR30 LITS-73 (FZ Yarway) Switch Fails Downscale

RR31 Reactor Recirc Suction Loop Rupture

RR32 Reactor Recirc Discharge Loop Rupture

RR33 Reference Leg Failure

RW01 Rod Worth Minimizer Failure

RW02 RSCS Failure

SL01 SLC Pump Trip

SL02 SLC Squib Valves Fail to Fire

SL03 SLC Discharge Relief Valve Fails Open

SW01 Service Water Pump Trip

SW02 Supply Strainer Blockage

SW03 Class - I Isolation

SW04 RHR Service Water Booster Pump Trip

SW05 Loss of REC to Recirc Pump

SW06 Service Water Leakage in Control Building Basement

SW07 TEC Pump Trip

SW08 TEC Heat Exchanger Tube Leak

SW09 REC Line Break

SW10 TEC Line Break

SW11 REC Pump Trip

SW12 REC Heat Exchanger Tube Leak

SW13 REC Heat Exchanger Cooling Water Blockage

SW14 REC Leakage in the Drywell

SW15 Loss of Drywell Cooling

SW16 PCIS Group 1 Isolation

TC01 Main Turbine Trip

TC02 Main Turbine Trip Lockout Relay Fails to Actuate

TC03 DEH Fluid Pump Unloader Valve Failure

TC04 DEH Oil System Pressure Oscillation

TC05 DEH Computer Failure

TC06 DEH System Speed Signal Failure

TC07 Bypass Valve Failure

TC08 Main Turbine Overspeed

TC09 Control Valve Failure

TC10 DEH Fluid Leakage

TC11 Turbine Main Stop Valve FAilure

TC12 DEH System Load Reference Failure

TU01 Main Turbine Bearing High Temperature

TU02 Main Turbine Bearing Oil Low Pressure

TU03 Main Turbine Bearing High Vibration

TU04 Main Turbine Oil Pump Failure

TU05 Main Turbine High Eccentricity

TU06 Main Turbine Rotor Expansion

TU07 Main Turbine Lift Oil Pump FAilure

YC01 PMIS Data Concentrator FAilure

Attachment C

SIMULATOR REMOTE FUNCTIONS

AD01 Drywell Pneumatic Supply

CS01 Cond Supply Vlv (CM-V-41) to CS A

CS02 Cond Supply Vlv (CMV-44) to CS B

CS03 CS PUmp A Suction CST Isol Vlv (CS-V-66)

CS04 CS Pump B Suction CST Isol Vlv (CS-V-67)

CU01 RWCU Pump 1A CRD Mini Purge PCV-FREG-1A

CU02 RWCU Pump 1B CRD Mini Purge PCV-FREG-1B

CU03 RWCU Filter/Demin FCV-15A Status

CU04 RWCU Filter/Demin FCV-15A Auto Setpoint

CU05 RWCU Filter/Demin FCV-15B Status

CU06 RWCU Filter/Demin FCV-15B Auto Setpoint

CU07 RWCU Pump 1A Isolation

CU08 RWCU Pump 1B Isolation

CU09 RR/CU Conductivity Sample Point Select

CU10 RWCU FCV-15A Manual Control

CU11 RWCU FCV-15B Manual Control

DG01 Loss of Emerg Trans; DG #1 & #2 Auto Start Boot

DG02 Loss of S/U Trans; DG #1 & #2 Auto Start Boot

DG03 #1 DG Droop/Parallel and Unit Parallel Switches

ED01 Electrical Lockout Relays Throughout Entire Plant

ED02 Ganged MCC Supply Brkrs with UV Trip (EXcept M,N,U,V)

ED03 250 VDC Turbine Building Starter Rack Transfer

ED04 125 VDC Panel "A" Transfer

ED05 125 VDC Panel "B" Transfer

ED06 125 VDC Reactor Building Starter Rack Transfer

ED07 125 VDC RCIC Starter Rack Transfer

ED08 125 VDC HPCI Starter Rack TRansfer

ED09 Critical Power Panel "CCP" Transfer

ED10 250 VDC BAttery Charger 1A in Service

ED11 125 VDC Battery Charger 1B in Service

ED12 125 VDC Battery Charger 1A in Service

ED13 125 VDC Battery Charger 1B in Service

ED14 24 VDC Battery Chargers A1 & A2 in Service

ED15 24 VDC Battery Chargers B1 & B2 in Service

ED16 460 V MCC "M" Supply Breaker Status

ED17 460 V MCC "N" Supply Breaker Status

ED18 460 V MCC "U" Supply Breaker Status

ED19 460 V MCC "V" Supply Breaker Status

ED20 Critical MCC "E" Power Supply Transfer Switch

ED21 Critical MCC "R" Power Supply Transfer Switch

ED22 Critical MCC "RA" Power Supply Transfer Switch

ED23 Critical MCC "RX" Power Supply Transfer Switch

ED24 Critical MCC "X" Power Supply Transfer Switch

ED25 125 VDC BAttery Charger IC in Service

ED26 125 VDC BAttery Charger IC Input Transfer Switch

ED27 125 VDC Battery Charger IC Output Transfer Switch

ED28 Gaitronics Power Supply Transfer Switch

ED29 "MAPP" Grid System Voltage

ED30 "MAPP" Grid System Frequency

EG01 Main Generator Disconnect

EG02 Main Generator Gas Pressure

EG03 Main Generator Hydrogen Purity

EG04 Hydrogen Seal Oil Backup Supply Pressure Regulation

FW01 Condensate Pump 1A Discharge Valve (MC-V-15)

FW02 Condensate Pump 1B Discharge Valve (MC-V-17)

FW03 Condensate Pump 1C Discharge Valve (MC-V-19)

FW04 Condensate Filter/Demin 1A Isolation Valves

FW05 Condensate Filter/Demin 1B Isolation Valves

FW06 Condensate Filter/Demin 1C Isolation Valves

FW07 Condensate Filter/Demin 1D Isolation Valves

FW08 Condensate Filter/Demin 1E Isolation Valves

FW09 Condensate Filter/Demin 1F Isolation Valves

FW10 Condensate F/D Bypass Valve

FW11 Condensate Flow Through AOG Condenser

FW12 Pressure Control Valve PCV-9A/B Setpoint

FW13 Condensate Booster Pump 1A Discharge Valve MO-301

FW14 Condensate Booster Pump 1B Discharge VAlve MO-302

FW15 Condensate Booster Pump 1C Discharge Valve MO-303

FW16 FW Heater 1A1 & 1A2 Isolation Valves MO-5/7

FW17 FW Heater 1A3 & 1A4 Isolation Valves MO-6/8

FW18 FW Heater 1A3 & 1A4 Isolation Valves MO-9/11

FW19 FW Heater 1B3 & 1B4 Isolation Valves MO-10/12

FW20 FW Heater 1A5 Isolation Valves MO-13/15

FW21 FW Heater 1B5 Isolation Valves MO-14/16

FW22 FW Heater Bypass Valve MO-17

FW23 FW Heater Bypass Valve MO-20

FW24 FW Heater Bypass Valve MO-21

FW25 FW Heater Bypass Valve MO-18

FW26 FW Heater Bypass Valve MO-22

FW27 FW Heater Bypass Valve MO-23

FW28 FW Heater Bypass Valve MO-19

FW29 Condensate Make-up & Reject to CST A or CST B

FW30 Condensate Recirc Pump Suction

FW31 Condensate Recirc PUmp

FW32 Condensate Recirc Pump Disch Valve FCV-13

FW33 FW Heater A5 to A4 Drain Valve (CD-MO-68)

FW34 FW HEater B5 to B4 Drain Valve (CD-MO-69)

FW35 FW HEater A3 to A2 Drain Valve (CD-MO-70)

FW36 FW HEater B3 to B2 Drain Valve (CD-MO-71)

FW37 CST Drain Valve to Main Condenser (MC-807)

FW38 ECST Makeup Valve

FW39 Condensate Transfer Pump & Disch Valve to Cond

FW40 Reactor Feed Startup Recirc Valves Isolation

FW41 Surge Dump Valve LCV-2A Isolation Valves

FW42 Surge Make-up Valve LCV-2B Isolation Valves

FW43 Condensate System and CBP Minimum Flow Isolation

FW44 RFP A Lovejoy Feedwater Demand Override

FW45 RFP B Lovejoy Feedwater Demand Override

HV01 Outside Air Temperature

HV02 Wind Speed

HV03 Wind Direction

HV04 Barometric Pressure

HV05 River Water Temperature

HV06 River Level (Sea Level Elev)

HV07 Outside RElative Humidity

IA01 Air Compressor Tendamatic Sequence Selection

IA02 PCV-609 Status

IA03 RHR, Core Spray Testable Check Valve Air Isolation

IA04 Air Compressor Reset

MC01 Main Condenser 1A1 Inlet and Outlet Valves

MC02 Main Condenser 1A2 Inlet and Outlet Valves

MC03 Main Condenser 1B1 Inlet and Outlet Valves

MC04 Main Condenser 1B2 Inlet and Outlet Valves

MC05 Main Condenser 1A2 Bypass Valve

MC06 Main Condenser 1A2 Bypass Valve

MC07 Main Condenser 1B1 Bypass Valve

MC08 Main Condenser 1B2 Bypass Valve

MC09 Main Condenser 1A Crossover Valve

MC10 Main Condenser 1B Crossover Valve

MC11 SJAE A In Service

MC12 SJAE B In Service

MC13 Deicing Gate Control

MS01 Auxiliary Boiler Status

MS02 Aux Steam Supply to RFPT (AS-MO-59)

MS03 Control for Extraction Steam NRV9 & NRV10

MS04 Control for Extraction Steam NRV3, NRV5 & NRV6

MS05 Control for Extraction Steam NRV11 & NRV12

MS06 Control for Extraction Steam NRV4, NRV7 & NRV8

NM01 RBM Channel A Auto Bypass Relay K7 Status

NM02 RBM Channel B Auto Bypass Relay K7 Status

OG01 Augmented Off-Gas System Charcoal Beds In Service

OG02 AOG Los Steam Pressure Isolation Relay

OG03 AOG Low Steam Pressure Isolation Signal Reset

PC01 Nitrogen Storage Tank Isolation Solenoid

PC02 Drywell/Torus N2 Supply (PC-13)

RC01 RCIC Isolation (IS) Switch (Aux Relay Room)

RD03 CRD PUmp Suction from CST/Demin Storage Rank

RD04 Charging Water Isolation Valve (CRD-29)

RD05 Isolation Valves for STabilizing Valves 25A

RD06 Isolation Valves for stabilizing Valves 25B

RD07 CRD PUmp A Discharge Valve (CRD-11)

RD08 CRD PUmp B Discharge Valve (CRD-12)

RD09 CRD Mini Purge to RR PUmp A

RD10 CRD Mini Purge to RR Pump B

RD11 Isolation Valves for Flow Control Valve 19A

RD12 Isolation Valves for flow Control Valve 19B

RD13 CRD Pumps Minimum Flow Valves

RD14 CRD PUmps Common Discharge Valve (CRD-170)

RD15 North and south LSDV Vent Valves

RD16 North SDV Drain Valve

RD17 South SDV Drain Valve

RH01 Cond Supply Vlvs (RH-V-103/104) to RHR A LPCI Line

RH02 Cond Supply Vlv (CM-V-38) to RHR B LPCI Line

RH03 RHR A Process Sample Vlv (SSV95/SSV96)

RH04 RHR B Process Sample Vlv (SSV6/SSV61)

RH05 RHR Pump A Suction from Cond Supply Vlv (RHR-98)

RH06 RHR Pump D Suction from Cond Supply Vlv (RHR-99)

RH07 RHR HEat EXchanger A Steam Isolation Vlv (RHR-368A)

RH08 RHR Heat Exchanger B Steam Isolation Vlv (RHR-368B)

RP01 RPS "A" MG Set Motor Start & Close Output Breaker

RP02 RPS :B: MG Set Motor Start & Close Output Breaker

RP03 Jumpers for LIS-101A/D Contacts AA2-AA3 and BB2-BB3

RP04 Jumpers for Contacts 9-41 (9-42) CC10-CC11 and CC12-CC13

RP05 Jumpers for Contacts EE12-EE13 and AA12-AA13 in Panel 9-15

RR01 Recirculation Pump A MG Set Lockout Relay

RR02 Recirculation Pump B MG Set Lockout Relay

RR03 RRMG A DC Oil Pump

RR04 RRMG B DC Oil Pump

RR05 RRMG A Scoop Tube

RR06 RRMG B Scoop Tube

SL01 SLC Suction Source

SW01 REC HXs SW Inlet Crosstie Vlv (SW-122)

SW02 REC HX1A Inlet Isol Vlv (REC-18)

SW03 REC HX1A Outlet Isol Vlv (REC-19)

SW04 REC HX1B Inlet Isol Vlv (REC-20)

SW05 REC HX1B Outlet Isol Vlv (REC-21)

SW06 RHSW Booster Pumps 1A, 1C Vlv to RH (SW-118)

SW07 RHSW Booster Pumps 1B, 1D Vlv to RH (SW-119)

SW08 RH HX1A SW Inlet Vlv (SW-145)

SW09 RH HX1B SW Inlet Vlv (SW-152)

SW10 Demin Water to REC Surge Tank

SW11 TEC HX1A SW Outlet Vlv (SW-55)

SW12 TEC HX1B SW Outlet Vlv (SW-64)

SW13 Demin Water to TEC Surge Tank

SW14 Air Compressor 1A Cooling Water

SW15 Air Compressor 1B Cooling Water

SW16 Air Compressor 1C Cooling Water

SW17 TEC HX1A Isolation Vlvs (TEC-18, 19)

SW18 TEC HX1B Isolation Vlvs (TEC-20. 21)

SW19 SW PUmps Disch Cross Connect Vlv (SW-36)

SW20 REC HX1A Temperature Control Vlv (SW-TCV-451A)

SW21 REC HX1B Temperature Control Vlv (SW-TCV-451B)

SW22 RRMG Set 1B Oil Cooling Vlv (REC-49)

SW23 RRMG Set 1B Oil Cooling Vlv (REC-51)

SW24 RHSW Boost Pumps Disch Vlv to RH (SW-120)

SW25 RHSW Booster Pump B Lifted Lead

SW26 Control Power Fuse for SW-MO89B

SW27 Local Operation of SW-MO-89B

TC01 Turbine Manual Trip from Front Standard

TU01 Turbine Lube Oil Cooler TEC Set Point (TIC-406)

TU02 Isolation for SW TCV to Turbine Lube Oil Cooler