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

U.S. NUCLEAR REGULATORY COMMISSION

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

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

SIMULATION FACILITY CERTIFICATION

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

FACILITY: Virgil C. Summer Nuclear Station

DOCKET NUMBER: 50-395

LICENSEE: South Carolina Electric & Gas Company

DATE: 3/22/91

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

Virgil C. Summer Nuclear Station
Nuclear Training Center
P.O. Box 88 Jenkinsville, SC 29065

[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)*

Refer to Table of Contents, Sections 3.1 through 3.4.

[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)*

Refer to Section 3.5 of the submittal, "Simulator Annual Operability Testing Requirements".

[ ] 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.

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

Not Applicable.

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: John L. Skolds

TITLE: Vice President, Nuclear Operations

DATE: 3/26/91

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

NRC FORM 474 (1-90)

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1.0 SIMULATOR INFORMATION

1.1 General

This report is being submitted as the initial certification report for the Virgil C. Summer Simulator. The Virgil C. Summer Simulator is the plant specific simulator for the Virgil C. Summer Nuclear Station, a Westinghouse 3-Loop Pressurized Water Reactor rated at 2775 MWT. The facility is owned by the South Carolina Electric & Gas Company and the South Carolina Public Service Authority, but operated solely by South Carolina Electric & Gas Company. The simulator was constructed by the Westinghouse Electric Corporation and was first utilized for training in February 1984.

Over the years, actual plant modifications were evaluated as to their impact on the simulator, and changes to the simulation system were made as deemed necessary. Consequently, the simulator has been maintained as an effective, accurate training and examination tool.

Commencing in 1990, a major simulator upgrade effort was undertaken and successfully completed in February 1991; the following features were incorporated as part of the upgrade:

- The SIMARC-4 package (Reactor Core, Reactor Coolant System and Steam Generator models)

- Improved Reactor Coolant System and Steam Generator Models: the reactor coolant system and steam generator models solve multinode, two phase, non-equilibrium thermodynamics conservation equations for two components (water and non-condensible gas). Separate equations are solved for mass (of each component and phase), for momentum, for energy (enthalpy of each region), and for pressure (volume). This allows for realistic simulation of normal and off normal conditions including mid-loop operations where the vessel has been partially drained. The reactor core model has been upgraded to provide a more realistic time varying calculation of the three dimensional power and temperature conditions in the reactor core. The time dependent

effects of xenon and samarium are treated in greater detail. The ability to adjust the kinetics response of the reactor to match design and/or plant data has been made simpler and easier. The SIMARC models also included the addition of plant components which were not previously simulated, allowing greater range of operability in training.

- The interactive Model Builder (IMB): the IMB was used to construct five new fluid models which replaced the existing models as part of the upgrade. These models are the Condensate and Main Feedwater System (CFW), the Chemical and Volume Control System (CVCS), the Component Cooling Water System (CCW), the Residual Heat Removal System (RHR), and the Safety Injection System (SIS). These models include plant modifications not heretofore incorporated into the simulation system, as well as improved solution techniques yielding models of higher order fidelity. SCE&G is in the process of reviewing the remaining twelve simulator fluid models for future upgrade using the IMB system.

  In addition, the IMB allows for the standardization of software thermal hydraulic models, provides more accurate documentation for these models, and produces the models in a structured program format. Finally, the IMB generated models have "Flexi-Leaks" capability. "Flexi-Leaks" permit the instructor to move system leak locations to any model node. Since model nodes occur at nearly all key locations in a system, an instructor is essentially unlimited in his choice of leak locations. "Flexi-Leaks" generate leaks which can represent true pipe shear with fluid exiting from both pipe ends.

- The Replacement Handler Package (RHP): the RHP includes Pump Handlers, Controller Handlers, Control Valve Handlers, Valve Handlers, Bistable Handlers, Transmitter Handlers, and Heat Exchanger Handlers. The benefits of these new handlers are three-fold. First, these new handlers include a variety of global failure options. These failures include one or more types of failures associated with each component simulated. This significantly

increases the number of abnormal or malfunction conditions which the instructor may initiate for training. Second, the handlers have been improved to provide a more accurate dynamic response of the components being simulated. Third, the initialization techniques employed by these handlers have been improved to allow for an easier and more thorough initialization procedure, resulting in a more accurate handler database.

- A Computerized Configuration Managemant System: a computer based Configuration Management System provides the capability of a single entry database. The hardware and all relevant software have been provided as part of this package.

- Generator and Transformer Electrical Relay Board Hardware and Software: the hardware and software required to simulate the Generator and Transformer Electrical Relay Board

1.2 Control Room Comparison

1.2.1 Control Room Physical Arrangement

The Virgil C. Summer Simulator physical arrangement closely duplicates the Virgil C. Summer Nuclear Station Control Room. The physical layout includes an artificial vertical column near the center of the control room in the same location as found in the plant control room. Emphasis is placed on maintaining the simulator appearance as close to the plant control room as possible in the area where normal, abnormal, and emergency operations of the plant would be conducted. The differences reported below have been evaluated not to have an impact upon training. Appendix 1 indicates the simulator control room areas that are controlled with regard to physical arrangement and environment.

Differences

1. A Simulator Instructor Booth is behind and above the Incore Panels in the controlled area.

2. The Tagging desk (including partitions, desks, chairs, and a computer terminal) is not included in the simulator controlled area.

3. Uncontrolled areas of the simulator are as indicated in Appendix 1. Generally, the uncontrolled area is behind the Main Control Board panels with the exceptions of the Heating, Ventilation, and Air Conditioning (HVAC) Panel; Generator and Transformer Electrical Relay Board; and Rod Disconnect Box. Also, the backs of the main control boards and the generator and relay board are uncovered for ease of maintenance and cooling.

1.2.2 Panels and Equipment

All major panels and equipment in the plant control room are present in the simulator control room with the following exceptions which are considered to have no impact on training or are not needed for training.

1. Integrated Plant Computer System (IPCS) terminal located on the north wall of the plant's control room (to be added at a later date)

2. IPCS color copier (to be added at a later date)

3. Nuclear Instrumentation Channel (NIS), NI-0033, spare NIS channel which is only statically simulated and not required for training.

4. Generator and Transformer Electrical Relay Board is statically simulated except for the following elements used for training:

    a. 86TTR Lockout Relay
    b. TS-412 Underfrequency Protection Switch
    c. Relay DC Power (Blue) lights.

5. Three (3) meteorological recorders normally mounted on back of main control board.

6. The simulator phone system does not have all of the capabilities of the plant phone system (such as speed dialing) but is adequate to support training needs and emergency plan drills and exercises.

7. Integrated Fire and Security (IFS) line printer and stand.

8. Environmental temperature monitor line printer.

9. Digital Metal Impact Monitor System (DMIMS) printer is statically simulated.

10. The Control Room Supervisor's desk and Shift Engineer's desk each contain two CRTs which are connected to the Simulator Integrated Plant Computer System at the simulator. Currently, the Plant Control Room contains one CRT on each desk driven from the Integrated Plant Computer System and one CRT on each desk which is driven from the Technical Support Center Computer. As soon as the Bypassed and Inoperable Status Indication (BISI) is approved for use on the Integrated Plant Computer System, the Plant Control Room will be modified to be consistent with the Simulator Control Room.

11. The following differences have also been noted on IST-9.1, *Simulator Control Board Audit*, Attachment I, Simulator Control Room Audit. These differences do not have an impact on training.

Panel 6106, 6107,	Item 2 - Simulator missing lower scale
Panel 6109,	Item 2 - Simulator has red band on scale
Panel 6109,	Item 5 - Simulator should have third scale
Panel 6111,	Item 6 - on FCV "MAN" is manual
Panel 6111,	Item 14 - "Push to Reset SG Blwdn Isol Vlv" should be raised
Panel 6112,	Item 4 - Left for channel select should be "←"
Panel 6115,	Item 10 - Alignment for Turbine Trip not correct
Panel NI-44(2),	Item 1 - Simulator has different timer scaler than plant
Panel RMS-3(5),	Item 2 - Plain bezel on Switch in Simulator Control Room
Panel RMS-3(5),	Item 3 - Switch has black gravoply with numbers (Simulator)
Panel RMS-3(5),	Item 5 - Same as Item 2, RMS-3(5)
Panel RMS-3(5),	Item 6 - Same as Item 3, RMS-3(5)
Panel RMS-3(5),	Item 8 - Same as Item 2, RMS-3(5)
Panel RMS-3(5),	Item 9 - Same as Item 3, RMS-3(5)
Panel RMS-2(5),	Item 2 - Plain bezel on Switch in Simulator Control Room
Panel RMS-2(5),	Item 3 - Simulator has black gravoply with numbers on edge

Panel RMS-2(5),	Item 5 - Same as Item 2, RMS-2(5)
Panel RMS-2(5),	Item 6 - Same as Item 3, RMS-2(5)
Panel RMS-2(5),	Item 8 - Same as Item 2, RMS-2(5)
Panel RMS-2(5),	Item 9 - Same as Item 3, RMS-2(5)
Panel RMS-1(3),	Item 1-10- Meter scale has 10^2, 10^3, 10^4 instead of 100, 1K, 10K
Panel FS-1(1),	Item 2 - Gaitronics phone station - Not In Simulator
Panel FS-1(1),	Item 6 - FS Printer #LA 75 Companion Printer - Not In Simulator
Panel FS-1(4),	Item 7 - FS Computer Printer Stand - Not In Simulator
Panel CCM-1(1),	Item 1 - Okidata, microline printer - Not In Simulator

1.2.3 Systems

All systems necessary to support operator training have been modeled on the simulator. The following is a list of systems which are either partially modeled or are not required to support control room operations.

BR	=	Boron Recycle (P)
CE	=	Control Room Evacuation (outside CR)
DN	=	Demin Water (P)
HR	=	HR Removal - Post Accident (P)
HY	=	Hydrogen (P)
MI	=	Miscellaneous Instruments (P)
NB	=	Nuclear Blowdown (P)
ND	=	Nuclear Plant Drains (P)
NG	=	Nitrogen (P)
OW	=	Off site Warning (P)
SD	=	Sump Drains (P)
SS	=	Nuclear sampling (P)
TX	=	Technical Support
VL	=	Local Ventilation (P)
WG	=	Waste Gas (P)
WI	=	Condensate Demin (P)

WL = Waste Liquid (P)

(P) = Partially modeled

1.2.4 Simulator Control Room Environment

The following simulator environmental differences are from IST-9.1, *Simulator Control Room Audit*. Appendix 1 defines the areas that the audit shall cover. Within the controlled area, emphasis is placed on maintaining the simulator control room like the plant control room.

1. Extra ceiling light to south of MCB
2. Two (2) upright cabinets in southwest corner
3. One (1) file cabinet below light box
4. Fire door in southwest corner
5. Insulated box above door
6. Printer facing wrong direction
7. Large electrical connector box on wall
8. Electrical connection box for printer
9. Electrical connection box
10. Plastic holders for forms
11. Wooden box for name plates
12. Boxing cover for fire door
13. No thermostat in simulator
14. No Radiation monitor in simulator
15. Different type of lights
16. No thermostat in simulator
17. Three (3) cameras in simulator
18. Lighting is different
19. Ceiling tiles are different
20. Simulator does not have access point
21. Doors are different
22. Simulator does not have phone
23. Simulator does not have doors to restrooms
24. No offices behind control board in simulator
25. Simulator does not have door in east wall

26. Ceiling has different size ceiling egg crates

1.2.5 Simulator Integrated Plant Computer System (SIPCS) Software, Point, and Display Differences

1. Appendix 2 of Section 1.2 contains the list of points which are "statically" presented by the simulator, and are available for display on the SIPCS. All points are assigned a value from the simulator, whether or not it is dynamically modeled, so that the SIPCS will have a value to display. The SIPCS database points are periodically updated to match the plant in accordance with simulator update requirements. All points listed in Appendix 2 have static (constant) values displayed.

2. The displays are updated to match the plant in accordance with simulator update requirements. All displays on the SIPCS contain the letters "SIM" in the lower right corner of the screen. Displays on the IPCS contain the letters "PLNT" in the lower right corner. This is a necessary human factors requirement since terminals in the TSC and EOF may be driven from either the SIPCS or the IPCS. (During periodic test drills, terminals in the TSC and EOF are driven by the simulator. During an actual emergency, these terminals would be driven by the plant IPCS.) In addition, for any displays which provide indication of voltage input for a point on the IPCS, that voltage is labeled 'NA for SIM' on the SIPCS.

3. The SIPCS gets its input data as engineering values from the simulator while the IPCS acquires data from a data acquisition subsystem as voltages. Only one data acquisition error quality on the SIPCS is simulated while the plant uses several different qualities so that data acquisition problems may be resolved more quickly.

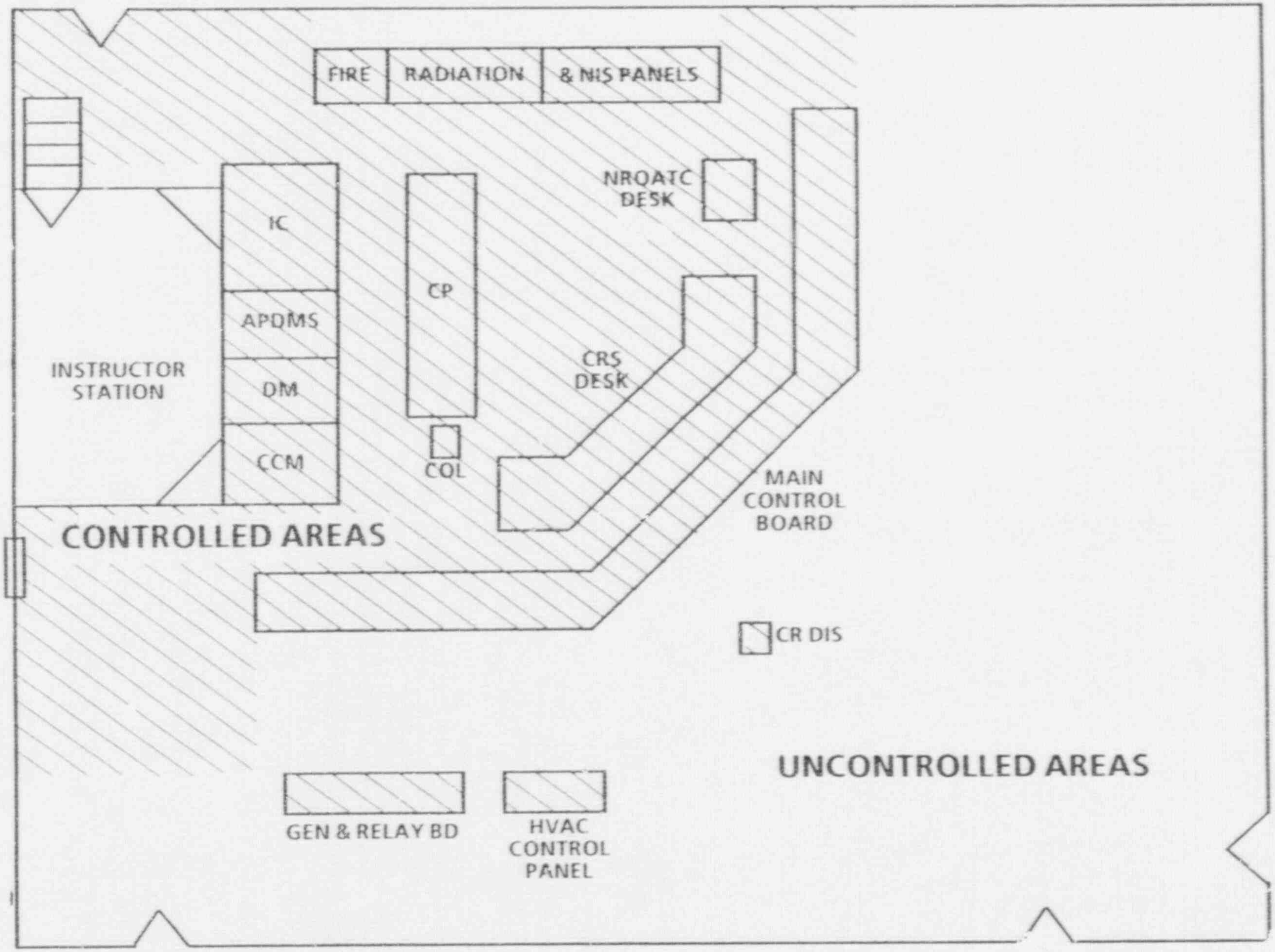
4. All reports from the SIPCS are labeled to indicate that they are simulator generated reports. Justification for the difference is a human factors concern.

5. The IPCS has six different scan classes ranging from 0.1 to 60 seconds per scan. The 0.1 second per scan class is not available for the SIPCS, leaving five possible scan classes.

6. The simulator unique functions of reset, freeze, and run were added to Applications Executive of the SIPCS. Some minor modifications to other software were also necessary to support these simulator commands.

7. All communications with the Main Control Board are processed via the simulator computer instead of directly controlled by the SIPCS. This includes control board annunciators and pen recorders.

SIMULATOR CONTROL ROOM CONTROLLED/UNCONTROLLED AREAS



STATICALLY MODELED SPDS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
T5764A	CCONA5	AI	THRUST BRG FRONT PLA	TE5764A
F4845A	CCON00	AI	NB SP RESIN SLUICE P	FT4845
F4811A	CCONA5	AI	NB DEMIN TO PLANT DI	FT4811
L4753A	CCON50	AI	NUCLEAR BLOWDOWN TAN	LT4753
L4843A	CCONA0	AI	SPENT RESIN STORAGE	LT4843
T3371A	CCONA5	AI	FPT A ACTIVE THRUST	TE3371
T3372A	CCONA0	AI	FPT B ACTIVE THRUST	TE3372
T3373A	CCONA5	AI	FPT C ACTIVE THRUST	TE3373
T3374A	CCONA5	AI	FPT A INACTIVE THRUS	TE3374
T3375A	CCONA0	AI	FPT B INACTIVE THRUS	TE3375
T3376A	CCONA5	AI	FPT C INACTIVE THRUS	TE3376
T3377A	CCON460	AI	FPT A HP STOP CHE	TE3377
T3378A	CCON460	AI	FPT B HP STOP VV CHE	TE3378
T3379A	CCON460	AI	FPT C HP STOP VV CHE	TE3379
T3384A	CCONA5	AI	FPT A OIL TEMP AT LP	TE3384
T3385A	CCONA0	AI	FPT B OIL TEMP AT LP	TE3385
T3386A	CCONA5	AI	FPT C OIL TEMP AT LP	TE3386
T3387A	CCONA5	AI	FPT A THRUST BRG DRA	TE3387
T3388A	CCONA5	AI	FPT B THRUST BRG DRA	TE3388
T3389A	CCONA0	AI	FPT C THRUST BRG DRA	TE3389
T3391A	CCONA5	AI	FPT A HP JOURN BRG D	TE3391
T3392A	CCONA5	AI	FPT B HP JOURN BRG D	TE3392
T3393A	CCONA5	AI	FPT C HP JOURN BRG D	TE3393
T3441A	CCONA5	AI	FW PUMP A INBD BRG T	TE3441
T3442A	CCONA5	AI	FW PUMP A OUTBD BRG	TE3442
T3443A	CCONA5	AI	FW PUMP A OUTER FACE	TE3443
T3491A	CCONA5	AI	FW PUMP A INNER FACE	TE3444
T3451A	CCONA5	AI	FW PUMP B INBD BRG T	TE3451
T3452A	CCONA5	AI	FW PUMP B OUTBD BRG	TE3452
T3453A	CCONA5	AI	FW PUMP B OUTER FACE	TE3453
T3494A	CCONA5	AI	FW PUMP B INNER FACE	TE3454
T3461A	CCONA5	AI	FW PUMP C INBD BRG T	TE3461
T3462A	CCONA5	AI	FW PUMP C OUTBD BRG	TE3462
T3463A	CCONA5	AI	FW PUMP C OUTER FACE	TE3463
T3497A	CCONA5	AI	FW PUMP C INNER FACE	TE3464
T3595A	CCON50	AI	EMER FW PUMP A MTR I	TE3595
T3596A	CCON50	AI	EMER FW PUMP A MTR O	TE3596
T3597A	CCON50	AI	EMER FW PUMP A MTR S	TE3597
T3605A	CCON50	AI	EMER FW PUMP B MTR I	TE3605
T3606A	CCON50	AI	EMER FW PUMP B MTR O	TE3606
T3607A	CCON50	AI	EMER FW PUMP B MTR S	TE3607
T4561A	CCONA5	AI	SERV WTR PP A MTR IN	TE4561
T4562A	CCONA5	AI	SERV WTR PP A MTR OU	TE4562
T4563A	CCONA0	AI	SERV WTR PP A MTR TH	TE4563
T4564A	CCONA0	AI	SERV WTR PP A MTR ST	TE4564
T4571A	CCONA5	AI	SERV WTR PP B MTR IN	TE4571
T4572A	CCONA5	AI	SERV WTR PP B MTR OU	TE4572
T4573A	CCONA0	AI	SERV WTR PP B MTR TH	TE4573
T4574A	CCONA0	AI	SERV WTR PP B MTR ST	TE4574
T4581A	CCONA0	AI	SERV WTR PP C MTR IN	TE4581

STATICALLY MODELED
SIPCS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
T4582A	CCONA0	AI	SERV WTR PP C MTR OU	TE4582
T4583A	CCONA0	AI	SERV WTR PP C MTR TH	TE4583
T4584A	CCON50	AI	SERV WTR PP C MTR ST	TE4584
T5763A	CCONA5	AI	THRUST BRG FRONT PLA	TE5763A
T5738A	CCONA5	AI	THRUST BRG REARPLA	TE5763B
T5737A	CCONA5	AI	THRUST BRG REARPLA	TE5764B
T5766A	CCON460	AI	1ST STAGE SHELL LWR	TE5766A
T5739A	CCON460	AI	1ST STAGE SHELL LWR	TE5766B
T5767A	CCON460	AI	MN TB CNTRL VV2 IN	TE5767A
T5740A	CCON460	AI	MN TB CNTRL VV2 OU	TE5767B
T5768A	CCON273	AI	MN TURBINESEAL STE	TE5768
T5775A	CCONA0	AI	MAIN TURBINE HYDRALL	TE5775
Y8412A	CCON36	AI	T.G. BUSHING COOLANT	TE5837
T7004A	CCON50	AI	COMP CLG PUMP A INBD	TE7004
T7005A	CCON50	AI	COMP CLG PUMP A OUTB	TE7005
T7006A	CCONA0	AI	COMP CLG PUMP A MTR	TE7006
T7007A	CCONA5	AI	COMP CLG PUMP A MTR	TE7007
T7008A	CCONA0	AI	COMP CLG PUMP A MTR	TE7008
T7014A	CCON50	AI	COMP CLG PUMP B INBD	TE7014
T7015A	CCON50	AI	COMP CLG PUMP B OUTB	TE7015
T7016A	CCON50	AI	COMP CLG PUMP B MTR	TE7016
T7017A	CCON50	AI	COMP CLG PUMP B MTR	TE7017
T7018A	CCON50	AI	COMP CLG PUMP B MTR	TE7018
T7024A	CCON50	AI	COMP CLG PUMP C INBD	TE7024
T7025A	CCON50	AI	COMP CLG PUMP C OUTB	TE7025
T7026A	CCON50	AI	COMP CLG PUMP C MTR	TE7026
T7027A	CCON50	AI	COMP CLG PUMP C MTR	TE7027
T7028A	CCON50	AI	COMP CLG PUMP C MTR	TE7028
T7381A	CCON50	AI	REAC BLD SP PP A INB	TE7381
T7382A	CCON50	AI	REAC BLD SP PP A OUT	TE7382
T7384A	CCON50	AI	REAC BLD SP PP A MTR	TE7384
T7385A	CCON50	AI	REAC BLD SP PP A MTR	TE7385
T7386A	CCON50	AI	REAC BLD SP PP A MTR	TE7386
T7391A	CCON50	AI	REAC BLD SP PP B INB	TE7391
T7392A	CCON50	AI	REAC BLD SP PP B OUT	TE7392
T7394A	CCON50	AI	REAC BLD SP PP B MTR	TE7394
T7395A	CCON50	AI	REAC BLD SP PP B MTR	TE7395
T7396A	CCON50	AI	REAC BLD SP PP B MTR	TE7396
T7557A	CCONA5	AI	CHG/SI PP A MTR INBD	TE7557
T7558A	CCONA5	AI	CHG/SI PP A MTR OUTB	TE7558
T7559A	CCON50	AI	CHG/SI PP A MTR STAT	TE7559
T7567A	CCONA0	AI	CHG/SI PP B MTR INBD	TE7567
T7568A	CCONA0	AI	CHG/SI PP B MTR OUTB	TE7568
T7569A	CCON50	AI	CHG/SI PP B MTR STAT	TE7569
T7577A	CCONA0	AI	CHG/SI PP C MTR INBD	TE7577
T7578A	CCONA0	AI	CHG/SI PP C MTR OUTB	TE7578
T7579A	CCON50	AI	CHG/SI PP C MTR STAT	TE7579
Y8101A	CCON50	AI	TB CHILLER OUTLET WA	TE9154
L1026	CCON50	AI	WASTE HOLDUP TANK LE	LT1001
L1027	CCON50	AI	CHEMICAL DRAIN TNK L	LT1002
L1030	CCON50	AI	LAUNDRY AND HOT SH T	LT1010

STATICALLY MODELED
SIPCS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
L1031	CCON50	AI	WASTE EVAP COND TNK	LT1012
L1032	CCON50	AI	FLOOR DRAIN TNK LEV	LT1077
L1033	CCON50	AI	WMT #1 LEVEL	ILT01082
L1034	CCON50	AI	WMT #2 LEVEL	ILT01803
P1014	CCON10	AI	GAS DECAY TANK 1 PFE	PT1036
P1015	CCON10	AI	GAS DECAY TANK 2 P[illegible]E	PT1037
P1016	CCON10	AI	GAS DECAY TANK 3 PREP	T1038
P1017	CCON10	AI	GAS DECAY TANK 4 PREP	T1039
P1018	CCON10	AI	GAS DECAY TANK 5 PREP	T1052
P1019	CCON10	AI	GAS DECAY TANK 6 PREP	T1053
P1020	CCON10	AI	GAS DECAY TANK 7 PREP	T1054
P1021	CCON10	AI	GAS DECAY TANK 8 PREP	T1055
DC01110H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC01110L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC01120H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC01120L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC01130H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC01130L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC011C0H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC011C0L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC01230H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC01230L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC01240H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC01240L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC01250H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC01250L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC012C0H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC012C0L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC03310H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC03310L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC03320H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC03320L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC03330H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC03330L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC03340H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC03340L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC2B1B0H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC2B1B0L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC2B2B0H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC2B2B0L	-CCON1[illegible]	AI	DALCAL VOLTS LOW CHA	
DC3311H	CCON	AI	DALCAL VOLTS HIGH CH	
DC3311L	-CCON	AI	DALCAL VOLTS LOW CHA	
DC3B1A0H	CCON[illegible]	AI	DALCAL VOLTS HIGH CH	
DC3B1A0L	-CCON10	AI	DALCAL VOLTS LOW CHA	
DC3B2A0H	CCON10	AI	DALCAL VOLTS HIGH CH	
DC3B2A0L	CCON10	AI	DALCAL VOLTS LOW CHA	
Y1803D	JTRUE	DI	MAIN STM ATMOSPHERIC	MNSTM-DUMPATMOS
Y2100D	JTRUE	DI	MSR A RS HI LOAD VAL	MSR-A-HILOADVLV
Y2101D	JTRUE	DI	MSR B RS HI LOAD VAL	MSR-B-HILOADVLV
Y2102D	JTRUE	DI	MSR A RS LO LOAD VAL	MSR-A-LOLOADVLV
Y2103D	JTRUE	DI	MSR B RS LO LOAD VAL	MSR-B-LOLOADVLV

STATICALLY MODELED
SIPCS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
Y2152D	JTRUE	DI	EHC FLUID RESERVOIR	EHC-FLUIDHIGH
Y2153D	JTRUE	DI	EHC FLUID RESERVOIR	EHC-FLUIDLOW
Y2200D	JTRUE	DI	MN TB CONTROL VLVS D	TB-CNTLVLVDMDO
Y2208D	JTRUE	DI	MAIN TURB THR BRG WE	TB-THRBRGWEAR
Y2212D	JTRUE	DI	MN TB VAC TRIP SYS B	TB-VACBELLOW
Y2213D	JTRUE	DI	EHC LOSS OF NEG 22 V	EHC-LOSS22V
Y2214D	JTRUE	DI	EHC LOSS OF POS 30 V	EHC-LOSS30V
Y2501D	JTRUE	DI	EHC LOSS OF 24 VDC	EHC-LOSS24V
Y2502D	JTRUE	DI	EHC SPEED SIGNALS LO	TBTR-EHCSPEED
Y2505D	JTRUE	DI	MOIS SEP HIGH LEVEL	BTR-MSHILV
Y2512D	JTRUE	DI	MAIN TURB BACKUP OVE	TBTR-BACKUPSPD
Y2727D	JTRUE	DI	GEN EMER SEAL OIL PP	TB-SEALOILAUTO
Y3241D	JTRUE	DI	FPT A ACTIVE THRUST	TPP0022A-PS2D
Y3242D	JTRUE	DI	FPT A INACTIVE THRUST	PP0022A-PS12D
Y3243D	JTRUE	DI	FPT B ACTIVE THRUST	TPP0022B-PS2D
Y3244D	JTRUE	DI	FPT B INACTIVE THRUS	TPP0022B-PS12D
Y3245D	JTRUE	DI	FPT C ACTIVE THRUST	TPP0022C-PS2D
Y3246D	JTRUE	DI	FPT C INACTIVE THRUS	TPP0022C-PS12D
Y3271D	JTRUE	DI	FPT A LP STOP VALVE	TPP0022A-LS1
Y3272D	JTRUE	DI	FPT B LP STOP VALVE	TPP0022B-LS1
Y3273D	JTRUE	DI	FPT C LP STOP VALVE	TPP0022C-LS1
Y3274D	JTRUE	DI	FPT A LP STOP VALVE	TPP0022A-LS
Y3275D	JTRUE	DI	FPT B LP STOP VALVE	PP0022B-LS9
Y3276D	JTRUE	DI	FPT C LP STOP VALVE	TPP0022C-LS9
Y3277D	JTRUE	DI	FPT A HP STOP VALVE	TPP0022A-LS2
Y3278D	JTRUE	DI	FPT B HP STOP VALVE	TPP0022B-LS2
Y3279D	JTRUE	DI	FPT C HP STOP VALVE	TPP0022C-LS2
Y9998D	JTRUE	DI	HALF SHELL CIRCUIT B	IPCS-CIRCBRKRS
DD1032	JTRUE	DI	EMRG TO 7200V 1DA BK	DAE1000-2
DD1002	JTRUE	DI	480V BUS 1DA1 BKR IN	DAF1000-2
DD1004	JTRUE	DI	480V BUS 1DA2 BKR IN	DAF1001-2
DD1030	JTRUE	DI	NORM TO 7200V 1DA BK	DAN1000-2
DD1033	JTRUE	DI	EMRG TO 7200V 1DB BK	DBE1000-2
DD1008	JTRUE	DI	480V BUS 1DB1 BKR IN	DBF1000-2
DD1010	JTRUE	DI	480V BUS 1DB2 BKR IN	DBF1001-2
DD1031	JTRUE	DI	NORM TO 7200V 1DB BK	DBN1000-2
DD1012	JTRUE	DI	DC CNTL PWR TO 7200V	DCL1000A-1
DD1013	JTRUE	DI	DC CNTL PWR TO 7200V	DCL1000B-1
DD1014	JTRUE	DI	DC CNTL PWR TO 480V	DCL2000A-1
DD1015	JTRUE	DI	DC CNTL PWR TO 480V	DCL2000B-1
DD1019	JTRUE	DI	DG A LOCAL CONTROL S	DGA-3
DD1020	JTRUE	DI	DG A READY FOR AUTO	DGA-4
DD1018	JTRUE	DI	DG A LOCAL TNFR SW I	DGA-5
DD1017	JTRUE	DI	DG A TO 7200V 1DA BK	DGA1000-2
DD1024	JTRUE	DI	DG B LOCAL CONTROL S	DGB-3
DD1025	JTRUE	DI	DG B READY FOR AUTO	DGB-4
DD1023	JTRUE	DI	DG B LOCAL TNFR SW I	DGB-5
DD1022	JTRUE	DI	DG B TO 7200V 1DB BK	DGB1000-2
ED1001	JTRUE	DI	7200V 1DA TO 1EA BKR	EAF1000-2
ED1003	JTRUE	DI	7200V 1DB TO 1EB BKR	EBF1000-2
ED1005	JTRUE	DI	480V BUS 1EA1 BKR IN	EF1000-2

STATICALLY MODELED SIPCS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
ED1007	JTRUE	DI	480V BUS 1EB1 BKR IN	EF1001-2
AD1073	JTRUE	DI	RHR HX BYP FLOW CNTL	FCV0605A-1
AD1074	JTRUE	DI	RHR HX BYP FLOW CNTL	FCV0605B-1
AD1091	JTRUE	DI	FW BYPASS CONT VLV	IFV3321-1
AD1092	JTRUE	DI	FW BYPASS CONT VLV	IFV3331-1
AD1093	JTRUE	DI	FW BYPASS CONT VLV	IFV3341-1
AD9500	JTRUE	DI	LETDOWN DIVERT TO RH	LCV115A-2
PD1000	JTRUE	DI	PRESSURIZER PWR RELI	PCV0444B
PD1001	JTRUE	DI	PRESSURIZER PWR RELI	PCV0445A
PD1002	JTRUE	DI	PRESSURIZER PWR RELI	PCV0445B
TD1000	JTRUE	DI	1DA1/1DA2 FDR BKR IN	TF1000-2
TD1001	JTRUE	DI	1DB1/1DB2 FDR BKR IN	TF1001-2
TD1002	JTRUE	DI	1EA1 FDR BKR IN TEST	TF1002-2
TD1003	JTRUE	DI	1EB1 FDR BKR IN TEST	TF1003-2
XD1025	JTRUE	DI	RB CLG FAN 64A BKR I	XFN0064A-2
XD1028	JTRUE	DI	RB CLG FAN 64B BKR I	XFN0064B-2
XD1030	JTRUE	DI	RB CLG FAN 65A BKR I	XFN0065A-2
XD1033	JTRUE	DI	RB CLG FAN 65B BKR I	XFN0065B-2
XD1036	JTRUE	DI	CHILLER A BKR IN TES	XHX0001A-2
XD1039	JTRUE	DI	CHILLER B BKR IN TES	XHX0001B-2
XD1042	JTRUE	DI	CHILLER C TR A BKR	XHX0001C-2
XD1149	JTRUE	DI	CHILLER C TR B BKR I	XHX0001C-7
Y7403D	JTRUE	DI	SPENT FUEL PURIFICA I	XPP0014
Y3501D-2	JTRUE	DI	MDEFW PUMP A BKR IN	XPP0021A-2
Y3502D-2	JTRUE	DI	MDEFW PUMP B BKR IN	XPP0021B-2
Y0600D-2	JTRUE	DI	RHR PUMP A BKR IN TE	XPP0031A-2
Y0601D-2	JTRUE	DI	RHR PUMP B BKR IN TE	XPP0031B-2
Y7351D-2	JTRUE	DI	RB SPRAY PUMP A BKR	XPP0038A-2
Y7352D-2	JTRUE	DI	RB SPRAY PUMP B BKR	XPP0038B-2
Y4704D-2	JTRUE	DI	SWBP A BKR IN TEST	XPP0045A-2
Y4705D-2	JTRUE	DI	SWBP B BKR IN TEST	XPP0045B-2
XD1108	JTRUE	DI	CHW PUMP A BKR IN TE	XPP0048A-2
XD1111	JTRUE	DI	CHW PUMP B BKR IN TE	XPP0048B-2
XD1114	JTRUE	DI	CHW PUMP C TR A BKR	XPP0048C-2
XD1151	JTRUE	DI	CHW PUMP C TR B BKR	XPP0048C-7
MD1035	JTRUE	DI	COND STOR TANK OUT B	XVG1007-1
Y2005D	JTRUE	DI	MN TB STOP VALVE 1 F	XVG2809A-SVOS
Y2006D	JTRUE	DI	MN TB STOP VALVE 2 F	XVG2809B-SVOS
Y2007D	JTRUE	DI	MN TB STOP VALVE 3 F	XVG2809C-SVOS
Y2008D	JTRUE	DI	MN TB STOP VALVE 4 F	XVG2809D-SVOS
Y2009D	JTRUE	DI	MN TB CONTROL VALVE	XVG2822A-C
Y2010D	JTRUE	DI	MN TB CONTROL VALVE	XVG2822B-C
Y2011D	JTRUE	DI	MN TB CONTROL VALVE	XVG2822C-C
Y2012D	JTRUE	DI	MN TB CONTROL VALVE	XVG2822D-C
Y2014D	JTRUE	DI	MN TB INTERMED STP V	XVM1462A-ISVCS
Y2022D	JTRUE	DI	MN TB INTERCEPT VALV	XVM1462A-IVCS
Y2015D	JTRUE	DI	MN TB INTERMED STP V	XVM1462B-ISVCS
Y2023D	JTRUE	DI	MN TB INTERCEPT VALV	XVM1462B-IVCS
Y2013D	JTRUE	DI	MN TB INTERMED STP V	XVM1463A-ISVCS
Y2021D	JTRUE	DI	MN TB INTERCEPT VALV	XVM1463A-IVCS
Y2016D	JTRUE	DI	MN TB INTERMED STP V	XVM1463B-ISVCS

STATICALLY MODELED
SIPCS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
Y2024D	JTRUE	DI	MN TB INTERCEPT VALV	XVM1463B-IVCS
Y1803DSE	JTRUE	SO	MAIN STM ATMOSPHERIC	MNSTMDUMPATMOSSE
Y2501DSE	JTRUE	SO	EHC LOSS OF 24 VDC	EHC-LOSS24VSE
Y2502DSE	JTRUE	SO	EHC SPEED SIGNALS LO	TBTR-EHCSPEEDSE
Y2505DSE	JTRUE	SO	MOIS SEP HIGH LEVEL	TBTR-MSHILVSE
DD1012SE	JTRUE	SO	DC CNTL PWR TO 7200V	DCL1000A-1SE
DD1013SE	JTRUE	SO	DC CNTL PWR TO 7200V	DCL1000B-1SE
DD1014SE	JTRUE	SO	DC CNTL PWR TO 480V	DCL2000A-1SE
DD1015SE	JTRUE	SO	DC CNTL PWR TO 480V	DCL2000B-1SE
PD1000SE	JTRUE	SO	PRESSURIZER PWR RELI	PCV0444BSE
PD1001SE	JTRUE	SO	PRESSURIZER PWR RELI	PCV0445ASE
PD1002SE	JTRUE	SO	PRESSURIZER PWR RELI	PCV0445BSE
Y0337D	JFALSE	DI	UNIT ON LINE TIE	UNIT-ON-LINE
Y2202D	JFALSE	DI	EHC LOAD LIMIT SET A	EHC-LOADLIMMAX
Y2518D	JFALSE	DI	LP HTR HI-HI LEVEL T	TBTR-LPHTRHIHIL
HD1000	JFALSE	DI	RHR HX FLOW CNTL	HCV0603A-1
HD1001	JFALSE	DI	RHR HX FLOW CNTL	HCV0603B-1
AD9011	JFALSE	DI	LETDOWN DIVERT TO VC	LCV115A-1
LD1010	JFALSE	DI	RHR PUMP ROOM SUMP	LS1966
LD1012	JFALSE	DI	RHR PUMP ROOM SUMP B	LS7702
LD1011	JFALSE	DI	AUX BLDG SUMP	LS7742
XD1000	JFALSE	DI	OUTSIDE AIR INTAKE I	XDP0003A-1
XD1001	JFALSE	DI	OUTSIDE AIR INTAKE I	XDP0003B-1
XD1002	JFALSE	DI	OUTSIDE AIR INTAKE I	XDP0004A-1
XD1003	JFALSE	DI	OUTSIDE AIR INTAKE I	XDP0004B-1
XD1004	JFALSE	DI	CR TO EQ ROOM ISOL	XDP0021A-1
XD1005	JFALSE	DI	CR TO EQ ROOM ISOL	XDP0021B-1
XD1006	JFALSE	DI	CR NORM AIR RETURN I	XDP0022A-1
XD1007	JFALSE	DI	CR NORM AIR RETURN I	XDP0022B-1
XD1008	JFALSE	DI	CR EMRG AIR UNIT INL	XDP0023A-1
XD1009	JFALSE	DI	CR EMRG AIR UNIT INL	XDP0023B-1
XD1010	JFALSE	DI	CR EMRG AIR UNIT OUT	XDP0024A-1
XD1011	JFALSE	DI	CR EMRG AIR UNIT OUT	XDP0024B-1
XD1012	JFALSE	DI	RB CLG FILTER BYP IS	XDP0110A-1
XD1013	JFALSE	DI	RB CLG FILTER BYP IS	XDP0110B-1
XD1014	JFALSE	DI	RB CLG FILTER BYP IS	XDP0111A-1
XD1015	JFALSE	DI	RB CLG FILTER BYP IS	XDP0111B-1
XD1016	JFALSE	DI	EQ ROOM RELIEF AIR E	XDP0133A-1
XD1017	JFALSE	DI	EQ ROOM RELIEF AIR E	XDP0133B-1
XD1018	JFALSE	DI	EQ ROOM RELIEF AIR E	XDP0234A-1
XD1019	JFALSE	DI	EQ ROOM RELIEF AIR E	XDP0234B-1
XD9015	JFALSE	DI	CR SUPPLY INLET ISOL	XDP0239A-1
XD9016	JFALSE	DI	CR SUPPLY INLET ISOL	XDP0239B-1
MD1036	JFALSE	DI	COND STOR TANK OUT	XVB1010-1
Y2018D	JFALSE	DI	MN TB INTRMD STP VLV	XVM1462A-ISVOS
Y2026D	JFALSE	DI	MN TB INTRCPT VALVE	XVM1462A-IVOS
Y2019D	JFALSE	DI	MN TB INTRMD STP VLV	XVM1462B-ISVOS
Y2027D	JFALSE	DI	MN TB INTRCPT VALVE	XVM1462B-IVOS
Y2017D	JFALSE	DI	MN TB INTRMD STP VLV	XVM1463A-ISVOS
Y2025D	JFALSE	DI	MN TB INTRCPT VALVE	XVM1463A-IVOS
Y2020D	JFALSE	DI	MN TB INTRMD STP VLV	XVM1463B-ISVOS

STATICALLY MODELED
SIPCS POINTS

POINT ID	VALUE	TYPE*	DESCRIPTION	INSTRUMENT
Y2028D	JFALSE	DI	MN TB INTRCPT VALVE	XVM1463B-IVOS
Y0198D	JFALSE	DI	BORON METER INVALID	BM-INVALID
Y0337DSE	JFALSE	SO	UNIT ON LINE TIE	UNIT-ON-LINESE
Y2512DSE	JFALSE	SO	MAIN TURB BACKUP OVE	TBTR-BACKUPSPDSE
Y2518DSE	JFALSE	SO	LP HTR HI-HI LEVEL T	TBTRLPHTHIHILSE

*Point Types
AI - Analog input
DI - Digital Input
SO - Sequence of Events Input (Digital)

1.3 Instructor Interface

1.3.1 Initial Conditions (ICs)

The Virgil C. Summer Simulator has the capacity for 78 Initialization Conditions (IC), of which 20 are protected from being changed by the use of an access code. Four Beginning of Life (BOL) ICs are available. They range from Hot Standby to 100% power. Thirteen Middle of Life (MOL) ICs range from Mid-loop Operation to 100% power. The three End of Life (EOL) ICs range from Hot Standby to 100%. Twenty ICs, not included in the 78, are used for Backtrack and Replay, which are saved every five minutes. Fifty-nine ICs may be used at the instructor's discretion. A list of the Initial Conditions can be found in Appendix 1

1.3.2 Malfunctions

The simulator is capable of simulating, in real time, abnormal and emergency events by inserting malfunctions. There are many malfunctions from which to choose. The degree of severity of the malfunction depends on the adjustable rate initiated by the instructor. The simulator malfunctions may be initiated with variable rates (with or without delay), by conditional response or by remote activation using a wireless, hand-held keypad. The malfunctions may be conveniently inserted and terminated. Also, several malfunctions may be initiated simultaneously, sequentially, or both. No matter which malfunctions are used or how they are initiated, the operator will not know that a problem exists except by indication on the control board. The malfunctions required by ANSI/ANS 3.5-1985 and supported by the Virgil C. Summer Simulator are listed in Appendix 2.

1.3.3 Controls Provided for Items Outside Control Room (LOAs)

The instructor has several means of providing controls that are required to operate items outside the control room. The majority of the actions performed outside the control room are accomplished by Local Operator Actions (LOA). Other control functions are the override and global handler features. By using these functions, the

instructor can perform all Local Operator Actions (LOAs) that are required by the Emergency Operating Procedures, Abnormal Operating Procedures and General Operating Procedures except the actions taken at the Control Room Evacuation Panel. If a plant or procedure change is made that requires a remote function which is not modeled, a Simulator Discrepancy Report is initiated, and the modeling is updated to accomplish the function. Appendix 3 is a listing of Local Operator Actions.

1.3.4 Additional Special Instructor/Training Features

1.3.4.1 Freeze - The simulator is capable of freezing simulation at any point without affecting the end state.

1.3.4.2 Simulator Speed - The speed of the simulation may be varied from zero to twice normal.

1.3.4.3 Backtrack - The backtrack function stores temporary snapshots at 5 minute intervals while the simulator is running, allowing the instructor the capability of stopping and going back (up to 100 minutes) to a previous point during the training evolution.

1.3.4.4 Replay - This feature allows the instructor to repeat exactly (in a semi-dynamic mode) all of the control board indications in the same sequence for the most recent operation (up to 100 minutes) performed on the simulator.

1.3.4.5 Snapshot - The instructor can store any present condition, in which simulation is frozen, to be recalled and used at a later time without any change in simulation.

1.3.4.6 Override - This allows the instructor to selectively override any of the switches/controllers, indicators, status lights, or annunciators used for control room indication.

1.3.4.7 Parameter Monitoring - This selection allows the instructor to monitor selected parameters. The parameter will update automatically at the selected frequency displaying the variable and its current value.

1.3.4.8 Graphic Trending - This selection allows the instructor to graphically monitor trends of up to four variables versus time using vector graphics.

1.3.4.9 X-Y Plot - This enables the instructor to generate a pictorial display of any selected variable with respect to any other variable.

1.3.4.10 Global Component Failures - These failures include one or more types of failures associated with each component simulated. This increases significantly the number of abnormal or malfunction conditions which the instructor may initiate for training. Global failure capabilities are:

Pumps
- Loss of Power
- Sheared Shaft
- Bearing failure to a variable extent
- Fail to auto start
- Trip on instructor command

Valves
- Leakage to a variable extent
- Fail as is
- Fail to a specified position (MOVs and AOVs)
- Fail open or close (solenoid valves)
- Fail due to loss of motive or control power

Heat exchangers
- Heat transfer fouling to a variable extent

Controllers

Fail to a selected position

Fail as is

Transducers

Fail to a selected value

Fail as is

Bistables

Actuated

Inhibited

Fail as is

1.3.4.11 "Flexi-Leaks" - This feature permits the instructor to move system leak locations to any upgraded model node. An instructor is essentially unlimited in his choice of leak locations. Up to three leaks per system can be inserted at any given time.

1.3.5 Simulator Operating Limits

An interface in the model coding for all affected models in the simulator is installed on the instructor station to notify the instructor that a simulator operating limit which is probable and which could significantly impact training has been exceeded.

Appendix 4 lists the Operating Limit Alarms that have been programmed into the simulator. When any of the limits are exceeded, a message identifying the limit (exactly as shown in the Appendix) will appear on the instructor console CRT. If more limits are subsequently exceeded, those messages will also appear on the CRT.

The primary objective of the operating limits alarm function is to provide a clear indication to the simulator instructor that further progress in the current operational evolution may result in negative training due to unrealistic simulation.

The limits are a combination of model limits and design limits. Model limits are limitations of the simulation codes which will, if exceeded, result in unrealistic simulation. These limits have been obtained from the modelers. The design limits were selected from safety related systems, systems which support safety related systems, or non-safety related systems whose failed components could either damage safety related systems or incur severe financial loss. Design limits were not considered if any of the following criteria were applicable:

- to reach the limit would require two or more inappropriate actions to be completed without instructor intervention,

- to reach the limit would require three or more failures or inappropriate actions to be completed,

- a simulated automatic response (which the instructor cannot or is unlikely to override via a system override or malfunction) will prevent reaching the limit.

- the system, component, etc., affecting the limit is not modeled,

- the limit cannot be reached by using functions available to the instructor.

Hence, the design limits selected are probable and they could have a significant impact on training.

Finally, the decision of whether or not to continue the training scenario once an operating limit has been exceeded rests entirely on the instructor and his training objectives. The Nuclear Operations Education and Training group sets up and evaluates training scenarios for inappropriate simulation prior to conducting the actual training. This practice will continue and is, in fact, the best method for avoiding negative training due to non-realistic simulation.

1.3.6 Monitoring Capabilities

The simulator is capable of monitoring up to 40 critical parameters with at least a 0.5 second resolution. Hard copy data is available, but the printing of data will slow simulation to an extent which is dependent on the print frequency and number of data points selected. In-plant computer printers located in the Control Room are driven by a separate computer and do not affect the simulation programs. The instructor system printer is not required during examinations or operator training.

AVAILABLE INITIAL CONDITION (IC) SETS

IC-1	-	MOL, Cold Shutdown, Secondary Plant Shutdown
IC-2	-	MOL, Mode 5, Going to Mode 4, 30 min. Prior to Bubble
IC-3	-	MOL, Mode 4, Going to Mode 5, Ready to Collapse Bubble
IC-4	-	MOL, Mode 3, Going to Mode 4, Prior to RHR
IC-5	-	MOL, Xe Free, Sm Peaked, Ready to Startup, MSIVS Shut
IC-6	-	MOL, 10-3% Power, Xe Free, Sm Peaked, MSIVS Shut
IC-7	-	MOL, 2% Power, Xe Free, Fw & Rev Flush, Turbine Warming
IC-8	-	MOL, 2% Power, Xe Free, Turbine Warming Complete Ready to Startup
IC-9	-	MOL, 15% Power, Ready to Startup Generator
IC-10	-	MOL, 100% Power, Equilibrium Xe, I & Sm
IC-11	-	MOL, 75% Power, Equilibrium Xe, I & Sm
IC-12	-	MOL, 50% Power, Equilibrium Xe, I & Sm
IC-13	-	MOL, 25% Power, Equilibrium Xe, I & Sm
IC-14	-	BOL, 100% Power, Equilibrium Xe, I & Sm
IC-15	-	BOL, Xe Free, Sm Peaked, Ready to Startup
IC-16	-	BOL, 75% Power, Equilibrium Xe, I & Sm
IC-17	-	EOL, 100% Power, Equilibrium Xe, I & Sm
IC-18	-	EOL, Xe Free, Sm Peaked, Ready to Startup
IC-19	-	EOL, 50% Power, After 3% per Min. Ramp Down from 100% Power
IC-20	-	MOL, Mid Loop Operations, 4" above Mid Loop, Manways Removed
IC-21 - 71		Instructor Snaps
IC-72 - 78		Simulator Operations Specialists Snaps
IC-79	-	Used with the Snap function
IC-80	-	Used with the Replay function
IC-81 - 100		Backup/Replay Snaps

REQUIRED MALFUNCTIONS

AUX-1	-	Loss of Instrument Air to All Headers
AUX-4	-	Service Water Pump Trip (Loss of Service Water System)
CCW-7	-	Component Cooling Water Pump Trip (Loss of CCW System)
CND-1	-	Loss of Condenser Vacuum
CND-2A	-	Hotwell Level Controller Failure (LC-3001) Hi
CRF-1	-	Rods Fail to Move
CRF-4	-	Dropped Rod
CRF-5	-	Ejected Rod
CRF-6	-	Uncontrolled Rod Motion
CRF-7	-	Stuck Rod
CVC-7	-	Loss of Normal Letdown
CVC-12	-	Leak in Charging Line
CVC-13	-	Letdown Line Leak Outside Containment
EPS-1	-	Station Blackout
EPS-3	-	Loss of Emergency Auxiliary Transformer
EPS-4	-	Loss of Service Buss
EPS-5	-	Loss of ESF Buss
EPS-6	-	Diesel Generator A Failure
EPS-7	-	Loss of 125VDC Bus 1HA
EPS-9	-	Loss of Unit Auxiliary Transformer
EPS-11	-	Loss of 120VAC Instrument Bus APN-5901
EPS-12	-	Generator Breaker Trips Without Turbine Trip
EPS-18	-	Failure of ESF Transformer
FWM-1	-	Main Feedwater Pump Trip
FWM-3	-	Emergency Feedwater Pump Trip
FWM-8	-	Feedline Break Inside Containment

REQUIRED MALFUNCTIONS

FWM-15A		FW Control Valve Position Failure (Open)
FWM-16A		FW Break Between FE 496 and FW Isol Valve 1611
MSS-3	-	Steamline Break Inside Containment
MSS-4	-	Steamline Break Outside Containment
MSS-5	-	Steam Dump Control Failure (Low)
MSS-6	-	Main Steam Isolation Valve Failure
MSS-10	-	Steam Generator Safety Valve Failure
NIS-3	-	Power Range Control Channel Failure
PCS-3	-	Steam Generator Control Failure
PCS-5	-	Safety Injection Failure (Inadvertent Initiation)
PCS-9	-	Reactor Trip Breaker Fails
PCS-14	-	Inadvertent Main Steam Isolation
PRS-1	-	Pressurizer Pressure Control Channel Failure (HI)
PRS-4	-	Relief Valve Failure (Interlock not Functional)
PRS-5	-	Pressurizer Pressure Channel Failure (Protection)
PRS-7	-	Safety Valve Failure
RCS-2	-	Steam Generator Tube Leak
RCS-3	-	Reactor Coolant Pump Trip
RCS-4	-	Reactor Coolant Pump Locked Rotor
RCS-5	-	Large LOCA in Cold Leg
RCS-6	-	Reactor Coolant System Leak
RCS-8	-	Reactor Coolant System RTD Failure
RCS-9	-	RCS Median Signal (Tavg or ΔT) Selector Failure
RCS-10	-	Reactor Coolant System Fuel Leak
RCS-15	-	Reactor Coolant System Flow Transmitter Failure
RCS-18	-	Variable RCS Boron Concentration
RHR-1	-	Residual Heat Removal Pump Trip

REQUIRED MALFUNCTIONS

RHR-7 - Reactor Building Spray Pump Failure

TUR-1 - Inadvertent Turbine Trip

LOCAL OPERATOR ACTION (LOA) LISTING

AUX1	SERVICE WATER PUMP C DISCONNECT SWITCH
AUX2	CHILLED WATER PUMP C DISCONNECT SWITCH
AUX3	SW PMP C ISOL TRAIN A
AUX4	SW PMP C ISOL TRAIN B
AUX5	HVAC CHILLER C ISOL TRAIN A
AUX6	HVAC CHILLER C ISOL TRAIN B
AUX7	LOA DOWNSTREAM OF HV8961
AUX8	PRESS REG FROM N2 SUPPLY
AUX9	HYDRO TEST PUMP DISCHARGE VALVE
AUX10	ACCUM N2 HEADER VALVE
AUX11	ISOL VALVE TO RCS HL 1
AUX12	ISOL VALVE TO RCS HL 2
AUX13	ISOL VALVE TO RCS HL 3
AUX14	ISOL VALVE TO RCS HL 1
AUX15	ISOL VALVE TO RCS HL 2
AUX16	ISOL VALVE TO RCS HL 3
AUX17	ISOL VALVE TO RCS CL 3
AUX18	ISOL VALVE TO RCS CL 2
AUX19	ISOL VALVE TO RCS CL 1
AUX20	ISOL VALVE TO RCS CL 3
AUX21	ISOL VALVE TO RCS CL 2
AUX22	ISOL VALVE TO RCS CL 1
AUX23	ACCUM A DRAIN TO RCDT
AUX24	ACCUM B DRAIN TO RCDT
AUX25	ACCUM C DRAIN TO RCDT
AUX26	BIT NETWORK BYPASS VALVE
AUX27	TEST LINE ISOL HEADER A
AUX28	TEST LINE ISOL HEADER B
AUX29	SPRAY PUMP COMMON TEST VALVE
AUX30	NAOH TANK OUTLET
AUX31	NAOH TANK DRAIN LINE
AUX32	NAOH TANK FILL LINE
AUX33	DSG CLR A ISO
AUX34	DSG CLR B ISO

LOCAL OPERATOR ACTION (LOA) LISTING

AUX35	SW TO CCW HX A ISO
AUX36	SW TO CCW HX B ISO
AUX37	HVAC CHLR A ISO
AUX38	HVAC CHLR B ISO
AUX39	SW BST PMP A SUCT
AUX40	SW BST PMP B SUCT
AUX41	HVAC CHLR C ISO A (3129A/B)
AUX42	HVAC CHLR C ISO B (3129C/D)
AUX43	HYDRO TEST PUMP SUPPLY BREAKER
AUX44	KITCHEN/TOILET EXH DMPR 245A
AUX45	KITCHEN/TOILET EXH DMPR 245B
AUX46	DEMIN WATER SYSTEM VACUUM DEGASIFIER CONTROL SWITCH
AUX47	NSFP32A SUCT ISOL VLV
AUX48	CROSS-TIE ISOL VLV
AUX49	NSFP32B SUCT ISOL VLV
AUX50	SFP HX 1B DISCH ISOL VLV
AUX51	SFP PMP 32B RTN TO SF POOL
AUX52	SFP TO RWST ISOL VLV
AUX53	SFP TO RWST ISOL VLV
AUX54	RB UNIT SELECTOR SW
AUX55	RB AIR COMP TR A MODE SEL SW
AUX56	RB AIR COMP TR B MODE SEL SW
AUX57	SS-9356A LOCAL CONTROL SWITCH
AUX58	SS-9356B LOCAL CONTROL SWITCH
AUX59	SS-9357 LOCAL CONTROL SWITCH
AUX60	SS-9364B LOCAL CONTROL SWITCH
AUX61	SS-9364C LOCAL CONTROL SWITCH
AUX62	SS-9365B LOCAL CONTROL SWITCH
AUX63	SS-9365C LOCAL CONTROL SWITCH
AUX64	SS-9387 LOCAL CONTROL SWITCH
AUX65	SS-9339 LOCAL CONTROL SWITCH
AUX66	SS-9341 LOCAL CONTROL SWITCH
AUX67	SS-9398A LOCAL CONTROL SWITCH
AUX68	SS-9398B LOCAL CONTROL SWITCH

LOCAL OPERATOR ACTION (LOA) LISTING

AUX69	SS-9398C LOCAL CONTROL SWITCH
AUX70	FIRE PROTECTION SYSTEM MOTOR PUMP OFF
AUX71	FIRE PROTECTION SYSTEM DIESEL PUMP OFF
AUX72	SPENT FUEL POOL PUMP A SWITCH
AUX73	SPENT FUEL POOL PUMP B SWITCH
AUX74	CELL SWITCH OF SERVICE WATER PUMP A
AUX75	CELL SWITCH OF SERVICE WATER PUMP B
AUX76	CELL SWITCH OF SERVICE WATER PUMP C TRAIN A
AUX77	CELL SWITCH OF SERVICE WATER PUMP C TRAIN B
AUX78	HVAC ANNUNCIATOR/HORN ENABLE
AUX79	RESET CONTROL RELAYS OF RB AIR COMPRESSOR AC/4A
AUX80	RESET CONTROL RELAYS OF RB AIR COMPRESSOR AC/4B
AUX81	RESET CONTROL RELAYS OF INSTRUMENT AIR COMPRESSOR AC/3A
AUX82	RESET CONTROL RELAYS OF INSTRUMENT AIR COMPRESSOR AC/3B
AUX83	RESET CONTROL RELAYS OF SUPPLEMENT AIR COMPRESSOR AC/12
AUX84	VALVE IA-2633, CONNECTS SUPPLEMENT AIR TO INSTRUMENT AIR
AUX85	SW PUMP A BKR - MANUAL TRIP
AUX86	SW PUMP B BKR - MANUAL TRIP
AUX87	SW PUMP C TR A BKR - MANUAL TRIP
AUX88	SW PUMP C TR B BKR - MANUAL TRIP
AUX89	SPRAY PUMP A BKR - MANUAL TRIP
AUX90	SPRAY PUMP B BKR - MANUAL TRIP
AUX91	HVAC CHLR PUMP A BREAKER STATUS
AUX92	HVAC CHLR PUMP B BREAKER STATUS
AUX93	HVAC CHLR PUMP C-A BREAKER STATUS
AUX94	HVAC CHLR PUMP C-B BREAKER STATUS
AUX95	HVAC CHLR A BREAKER STATUS
AUX96	HVAC CHLR B BREAKER STATUS
AUX97	HVAC CHLR C-A BREAKER STATUS
AUX98	HVAC CHLR C-B BREAKER STATUS
AUX99	HVAC CHLR A RESET
AUX100	HVAC CHLR B RESET
AUX101	HVAC CHLR C RESET
AUX102	RBCU INLET VLV 3108A BREAKER STATUS

LOCAL OPERATOR ACTION (LOA) LISTING

AUX103	RBCU INLET VLV 3108B BREAKER STATUS
AUX104	RBCU INLET VLV 3108C BREAKER STATUS
AUX105	RBCU INLET VLV 3108D BREAKER STATUS
AUX106	RBCU OUTLET VLV 3109A BREAKER STATUS
AUX107	RBCU OUTLET VLV 3109B BREAKER STATUS
AUX108	RBCU OUTLET VLV 3109C BREAKER STATUS
AUX109	RBCU OUTLET VLV 3109D BREAKER STATUS
AUX110	DIESEL AIR CMPR TO IA SUPPLY VALVE - XVG 2670
AUX111	CELL SWITCH FOR CRDM FAN 67A
AUX112	CELL SWITCH FOR CRDM FAN 67B
AUX113	VALVE STEM LEAKOFF ANNUN RESET SWITCH
CCW1	CCW PUMP C DISCONNECT SWITCH
CCW2	CC SRG TK A SIDE DRN
CCW3	CC HX A DISCH ISO VLV
CCW4	CC HX B DISCH ISO VLV
CCW5	CC HX A INLET ISO VLV
CCW6	CC HX B INLET ISO VLV
CCW7	CC FROM RHR HX A ISO VLV
CCW8	CC FROM RHR HX B ISO VLV
CCW9	CC PP C SUCT LP B ISO VLV
CCW10	CC PP C SUCT LP A ISO VLV
CCW11	CC PP C DISCH LP B ISO VLV
CCW12	CC PP C DISCH LP A ISO VLV
CCW13	CC TO LTDN HX ISO VLV
CCW14	CC TO SL WTR HX ISO VLV
CCW15	CC-V7096 CCW SRG TNK VENT VLV
CCW16	CC-V9557 DEMIN WTR TO CCW SRG TK
CCW17	CC-9547A CCW MKUP TO SRG TK
CCW18	CC-9547B CCW MKUP TO SRG TK
CCW19	CC-9624A CCW TO SFP HX A
CCW20	CC-9624B CCW TO SFP HX B
CCW21	CC-9621A CCW TO WST GAS CMPSR A
CCW22	CC-9621B CCW TO WST GAS CMPSR B
CCW23	CC-9661A CCW TO H2 CAT RECOMB A

LOCAL OPERATOR ACTION (LOA) LISTING

CCW24	CC-9661B CCW TO H2 CAT RECOMB B
CCW25	CC-9615 LTDN HX OUTLET ISO
CCW26	CC-9620 SL WTR HX OUTLET ISO
CCW27	CC-9597A CCW BST PP A DISCH
CCW28	CC-9597B CCW BST PP B DISCH
CCW29	CC-9597C CCW BST PP C DISCH
CCW30	CC-9590A CCW TO RCP A THERM BARR
CCW31	CC-9590B CCW TO RCP B THERM BARR
CCW32	CC-9590C CCW TO RCP C THERM BARR
CCW33	CC-9586A CCW TO RCP A UPPER BRG
CCW34	CC-9586B CCW TO RCP B UPPER BRG
CCW35	CC-9586C CCW TO RCP C UPPER BRG
CCW36	CC-9587A CCW TO RCP A LOWER BRG
CCW37	CC-9587B CCW TO RCP B LOWER BRG
CCW38	CC-9587C CCW TO RCP C LOWER BRG
CCW39	CC-9501A CCW PP A DISCH ISO
CCW40	CC-9501B CCW PP B DISCH ISO
CCW41	CC-9501C CCW PP C DISCH ISO
CCW42	CELL SWITCH OF CCW PUMP A
CCW43	CELL SWITCH OF CCW PUMP B
CCW44	CELL SWITCH OF CCW PUMP C TRAIN A
CCW45	CELL SWITCH OF CCW PUMP C TRAIN B
CCW46	CCW PUMP A BKR - MANUAL TRIP
CCW47	CCW PUMP B BKR - MANUAL TRIP
CCW48	CCW PUMP C TR A BKR - MANUAL TRIP
CCW49	CCW PUMP C TR B BKR - MANUAL TRIP
CCW50	CCW PUMP A MANUAL SPEED SELECTOR SWITCH
CCW51	CCW PUMP B MANUAL SPEED SELECTOR SWITCH
CCW52	CCW PUMP C MANUAL SPEED SELECTOR SWITCH
CND1	CW - LP CND CW INLET ISOL. VLV 806A
CND2	CW - LP CND CW INLET ISOL. VLV 806B
CND3	CW - AUX CND A CW ISOL. VLV 820A
CND4	CW - AUX CND B CW ISOL. VLV 820B
CND5	CW - AUX CND C CW ISOL. VLV 820C

LOCAL OPERATOR ACTION (LOA) LISTING

CND6	CW - BYPASS OCBP LINE ISOL. VLV 863
CND7	CW - H2 COOLER DISCH. ISOL. VLV 869B
CND8	CW - H2 COOLER TCV BYPASS ISOL. VLV 4265B
CND9	CW - LUBE OIL COOLER TCV BYPASS ISOL. VLV 881
CND10	CW - MAIN CNDSR INLET WATERBOX DRAIN VLV 171A
CND11	CW - MAIN CNDSR INLET WATERBOX DRAIN VLV 171B
CND12	CW - MAIN CNDSR OUTLET WATERBOX DRAIN VLV 172A
CND13	CW - MAIN CNDSR OUTLET WATERBOX DRAIN VLV 172B
CND14	CW - MAIN CNDSR INLET WATERBOX DRAIN VLV 173A
CND15	CW - MAIN CNDSR INLET WATERBOX DRAIN VLV 173B
CND16	CW - MAIN CNDSR OUTLET WATERBOX DRAIN VLV 174A
CND17	CW - MAIN CNDSR OUTLET WATERBOX DRAIN VLV 174B
CND18	CW - MAIN CNDSR INLET WATERBOX VENT VLV 162
CND19	CW - MAIN CNDSR INLET WATERBOX VENT VLV 163
CND20	CW - MAIN CNDSR OUTLET WATERBOX VENT VLV 164
CND21	CW - MAIN CNDSR OUTLET WATERBOX VENT VLV 165
CND22	CW - MAIN CNDSR INLET WATERBOX VENT VLV 175
CND23	CW - MAIN CNDSR INLET WATERBOX VENT VLV 176
CND24	CW - MAIN CNDSR OUTLET WATERBOX VENT VLV 166
CND25	CW - MAIN CNDSR OUTLET WATERBOX VENT VLV 167
CND26	CW - CW VLV 807A SWITCH POSITION
CND27	CW - CW VLV 807B SWITCH POSITION
CND28	COND PP A DISCH VLV XVB-614A HAND SWITCH
CND29	COND PP B DISCH VLV XVB-614B HAND SWITCH
CND30	COND PP C DISCH VLV XVB-614C HAND SWITCH
CND31	CC- V 631 CST FILL VLV ISOL VLV
CND32	CC- V 633 CST FILL VLV BYPASS VLV
CND33	CC- V 650 COND HOTWELL NORM MU ISOL VLV
CND34	CC- V 652 COND HOTWELL NORM MU BYPASS VLV
CND35	CC- V 653 COND HOTWELL EMER MU ISOL VLV
CND36	CC- V 655 COND HOTWELL EMER MU BYPASS VLV
CND37	CONDENSER AIR INLEAKAGE RATE (SCFM)
CND38	COND POLISHING SYS HIGH DP RESET PB
CND39	COND POLISHING SYS BYPASS VALVE SWITCH POSITION

LOCAL OPERATOR ACTION (LOA) LISTING

CND40	COND POLISHING FLOW CONTROL STATION A DEMAND
CND41	COND POLISHING FLOW CONTROL STATION B DEMAND
CND42	COND POLISHING FLOW CONTROL STATION C DEMAND
CND43	AUX. COND. VACUUM BREAKERS AR-123 A/B/C
CND44	COND TO S/G BD TC-3062A AUTO-MANUAL MODE SELECTOR
CND45	COND TO S/G BD TC-3062B AUTO-MANUAL MODE SELECTOR
CND46	COND TO S/G BD TC-3062C AUTO-MANUAL MODE SELECTOR
CND47	COND TO S/G BD TV-3062A MANUAL POSITION
CND48	COND TO S/G BD TV-3062B MANUAL POSITION
CND49	COND TO S/G BD TV-3062C MANUAL POSITION
CND50	OPEN CYCLE BOOSTER PUMP A BKR - MANUAL TRIP
CND51	OPEN CYCLE BOOSTER PUMP B BKR - MANUAL TRIP
CND52	CONDENSATE PUMP A BKR - MANUAL TRIP
CND53	CONDENSATE PUMP B BKR - MANUAL TRIP
CND54	CONDENSATE PUMP C BKR - MANUAL TRIP
CND55	CIRC WATER PUMP A BKR - MANUAL TRIP
CND56	CIRC WATER PUMP B BKR - MANUAL TRIP
CND57	CIRC WATER PUMP C BKR - MANUAL TRIP
CVC1	V 8403 - FV 122 BYPASS VALVE
CVC2	V 8402B - FV 122 ISOL VALVE
CVC3	V 8441 - EMERGENCY DILUTION
CVC4	V 8439 - MAKEUP TO PUMPS
CVC5	V 8432 - RWST + MAKEUP ISOL
CVC6	V 8434 - RWST + MAKEUP ISOL
CVC7	V 8419 - VCT DRAIN
CVC8	V 8376 - N2 SUPPLY ISOL
CVC9	V 7921 - N2 SUPPLY ISOL
CVC10	V 8410 - H2 SUPPLY ISOL
CVC11	V 8323A - BA TANK SUCTION ISOL
CVC12	V 8323B - BA TANK SUCTION ISOL
CVC13	V 8320A - MINIFLOW ISOL
CVC14	V 8320B - MINIFLOW ISOL
CVC15	V 8321A - EXTRA RECIRC ISOL
CVC16	V 8321B - EXTRA RECIRC ISOL

LOCAL OPERATOR ACTION (LOA) LISTING

CVC17	V 8301A BRS CROSSTIE VALVE
CVC18	V 8301B BRS CROSSTIE VALVE
CVC19	V 8311A BATCHING CROSSTIE VALVE
CVC20	V 8311B BATCHING CROSSTIE VALVE
CVC21	V 8315A PUMP DISCH ISOL
CVC22	V 8315B PUMP DISCH ISOL
CVC23	V 8569 - BRS BA REFILL ISOL
CVC24	V 8308-BATCHING TANK ISOL
CVC25	V 8302- MAKEUP TO BATCH TANK
CVC26	V 8408A PV 145 ISOLATION
CVC27	V 8409- PV 145 BYPASS
CVC28	V 8516- CATION DEMIN ISOL
CVC29	V 8524A MIXED DEMIN ISOL
CVC30	V 8524B MIXED DEMIN ISOL
CVC31	V 8514- CATION DEMIN BYPASS
CVC32	V 8384A SW INJ FILTER A ISOL
CVC33	V 8384B SW INJ FILTER B ISOL
CVC34	V 8484 SW RTN TO CHG PUMPS
CVC35	V 8482 SW RTN TO VCT
CVC36	SYSTEM LEAK RATE TO CONTAINMENT
CVC37	V 8389 HV 186 BYPASS VALVE
CVC38	V 8369A - SEAL INJECTION THROTTLE VALVE
CVC39	V 8369B - SEAL INJECTION THROTTLE VALVE
CVC40	V 8369C - SEAL INJECTION THROTTLE VALVE
CVC41	CHARGING PUMP A SUPPLY BRKR
CVC42	CHARGING PUMP B SUPPLY BRKR
CVC43	CHARGING PUMP C SUPPLY BRKR TRAIN A
CVC44	CHARGING PUMP C SUPPLY BKR TRAIN B
CVC45	CHARGING PUMP C DISCONNECT SWITCH
CVC46	PCV-1092 LOW VCT PRESSURE CLOSURE RESET
CVC47	HCV-1094 VCT TO GWPS FLOW SETPOINT
CVC48	CHARGING PUMP A BKR - MANUAL TRIP
CVC49	CHARGING PUMP B BKR - MANUAL TRIP
CVC50	CHARGING PUMP C TR A BKR - MANUAL TRIP

LOCAL OPERATOR ACTION (LOA) LISTING

CVC51	CHARGING PUMP C TR B BKR - MANUAL TRIP
CVC52	CHARGING SYS LEAK TO WEST PENETRATION AREA
CVC53	ISOLATION VALVE 7010D BTRS DEMIN D
CVC54	VENT CHARGING PUMP A AFTER CAVITATION
CVC55	VENT CHARGING PUMP B AFTER CAVITATION
CVC56	VENT CHARGING PUMP C AFTER CAVITATION
CVC57	V 8430 - BA BLENDER INLET ISOLATION
CRF1	P/A CONVERTER
CRF2	MANUAL ROD SPEED SET POINT
CRF3	MG SET A BREAKER STATUS
CRF4	MG SET B BREAKER STATUS
EPS1	EPS BREAKER 0/C TRIP RESET
EPS2	MAIN GEN MASTER RESET
EPS3	7.2KV BUS A DISCONNECT SWITCH
EPS4	7.2KV BUS B DISCONNECT SWITCH
EPS5	7.2KV BUS C DISCONNECT SWITCH
EPS6	7.2KV BUS DB DISCONNECT SWITCH
EPS7	ES XFMR 1 DISCONNECT
EPS8	ES XFMR 2 DISCONNECT BREAKER
EPS9	LOCAL BREAKER A13C
EPS10	LOCAL BREAKER C14C
EPS11	SELECT TRAIN FOR MCC 1EC1X
EPS12	SELECT TRAIN FOR BATT CGR 1A/1B
EPS13	OPERATE MCC 1A1X ATU
EPS14	OPERATE MCC 1A4X ATU
EPS15	OPERATE MCC 1A4Y ATU
EPS16	OPERATE MCC 1B1X ATU
EPS17	OPERATE MCC 1C1X ATU
EPS18	BATT CGR 1A INPUT BREAKER
EPS19	BATT CGR 1B INPUT BREAKER
EPS20	BATT CGR 1X INPUT BREAKER
EPS21	BATT CGR 1A OUTPUT BREAKER
EPS22	BATT CGR 1B OUTPUT BREAKER
EPS23	BATT CGR 1X OUTPUT BREAKER

LOCAL OPERATOR ACTION (LOA) LISTING

EPS24	BATT CGR 1X/2X INPUT BREAKER
EPS25	BATT CGR 1X/2X OUTPUT BREAKER
EPS26	BATT 1A OUTPUT BREAKER
EPS27	BATT 1B OUTPUT BREAKER
EPS28	BATT 1X OUTPUT BREAKER
EPS29	480V MCC 1A2X
EPS30	480V MCC 1A3X LOCAL BREAKER
EPS31	480V MCC 1A3Y LOCAL BREAKER
EPS32	480V MCC 1B1Y LOCAL BREAKER
EPS33	480V MCC 1B2X LOCAL BREAKER
EPS34	480V MCC 1B3X LOCAL BREAKER
EPS35	480V MCC 1B3Y LOCAL BREAKER
EPS36	480V MCC 1B4X LOCAL BREAKER
EPS37	480V MCC 1C2X LOCAL BREAKER
EPS38	480V MCC 1C3X LOCAL BREAKER
EPS39	480V. MCC 1C3Y LOCAL BREAKER
EPS40	480V MCC 1C4X LOCAL BREAKER
EPS41	480V MCC 1DA2X LOCAL BREAKER
EPS42	480V MCC 1DA2Y LOCAL BREAKER
EPS43	480V MCC 1DA2Z LOCAL BREAKER
EPS44	480V MCC 1DB2X LOCAL BREAKER
EPS45	480V MCC 1DB2Y LOCAL BREAKER
EPS46	480V MCC 1DB2Z LOCAL BREAKER
EPS47	480V MCC 1EA1X LOCAL BREAKER
EPS48	480V MCC 1EB1X LOCAL BREAKER
EPS49	480V MCC 1EC1X LOCAL BREAKER
EPS50	120VAC BUS 1FA
EPS51	120VAC BUS 1FB
EPS52	120VAC BUS 1FX
EPS53	125VDC BUS 1HA1
EPS54	125VDC BUS 1HA2
EPS55	125VDC BUS 1HA3
EPS56	125VDC BUS 1HB1
EPS57	125VDC BUS 1HB2

LOCAL OPERATOR ACTION (LOA) LISTING

EPS58	125VDC BUS 1HB3
EPS59	125VDC BUS 1HX1
EPS60	125VDC BUS 1HX2
EPS61	125VDC BUS 1HX3
EPS62	INVERTER 1 AC/DC INPUT BREAKER
EPS63	INVERTER 2 AC/DC INPUT BREAKER
EPS64	INVERTER 3 AC/DC INPUT BREAKER
EPS65	INVERTER 4 AC/DC INPUT BREAKER
EPS66	INVERTER 5 AC/DC INPUT BREAKER
EPS67	INVERTER 6 AC/DC INPUT BREAKER
EPS68	INVERTER 7 AC/DC INPUT BREAKER
EPS69	INVERTER 8 AC/DC INPUT BREAKER
EPS70	INVERTER 7 TERM BLOCK
EPS71	INVERTER 8 TERM BLOCK
EPS72	120VAC VITAL INST SOURCE SELECT #1
EPS73	120VAC VITAL INST SOURCE SELECT #2
EPS74	120VAC VITAL INST SOURCE SELECT #3
EPS75	120VAC VITAL INST SOURCE SELECT #4
EPS76	120VAC VITAL INST SOURCE SELECT #5
EPS77	120VAC VITAL INST SOURCE SELECT #6
EPS78	120VAC VITAL INST SOURCE SELECT #7
EPS79	120VAC VITAL INST SOURCE SELECT #8
EPS80	SWD BUS TRIP RESET
EPS81	SWD BKR 8852 SCPSA #1
EPS82	SWD BKR 8842 PINELAND #1
EPS83	SWD BKR 8832 DENNY TERRACE #2
EPS84	SWD BKR 8792 DENNY TERRACE #1
EPS85	SWD BKR 8772 PARR #2
EPS86	SWD BKR 8942 FAIRFIELD #1
EPS87	SWD BKR 8912 FAIRFIELD #2
EPS88	SWD BKR 8742 PARR #1
EPS89	SWD BKR 8732 GRANITEVILLE
EPS90	SWD BKR 8722 SCPSA #2
EPS91	SWD BKR 8822 BUS 1-3 TIE

LOCAL OPERATOR ACTION (LOA) LISTING

EPS92	SWD 115KV LINE
EPS93	DG A GOV CNTRL PWR BLOWN FUSE INDICATOR
EPS94	DG A LOCAL MODE
EPS95	DG A LOCAL TEST START SW
EPS96	DG A LOCAL TEST STOP SW
EPS97	DG A LOCAL EMER START SW
EPS98	DG A LOCAL EMER START RESET PB
EPS99	DG A LOCAL SPEED RAISE SW
EPS100	DG A LOCAL SPEED LOWER SW
EPS101	DG A LOCAL VOLTAGE RAISE SW
EPS102	DG A LOCAL VOLTAGE LOWER SW
EPS103	DG A LOCAL VOLT REG MAN MODE SW
EPS104	DG A LOCAL FIELD RAISE SW
EPS105	DG A LOCAL FIELD LOWER SW
EPS106	DG A LOCAL ENG SHTDN RESET SW
EPS107	DG A LOCAL GEN SHTDN RESET SW
EPS108	DG A LOCAL EXCITER SHTDN SW
EPS109	DG A LOCAL EXCITER SHTDN RESET SW
EPS110	DG A LOCAL MANUAL FIELD FLASH SW
EPS111	CLOSE DG A BKR TO BUS DA (LOCAL)
EPS112	TRIP DG A BKR TO BUS DA (LOCAL)
EPS113	DG B GOV CNTRL PWR BLOWN FUSE INDICATOR
EPS114	DG B LOCAL MODE
EPS115	DG B LOCAL TEST START SW
EPS116	DG B LOCAL TEST STOP SW
EPS117	DG B LOCAL EMER START SW
EPS118	DG B LOCAL EMER START RESET PB
EPS119	DG B LOCAL SPEED RAISE SW
EPS120	DG B LOCAL SPEED LOWER SW
EPS121	DG B LOCAL VOLTAGE RAISE SW
EPS122	DG B LOCAL VOLTAGE LOWER SW
EPS123	DG B LOCAL VOLT REG MAN MODE SW
EPS124	DG B LOCAL FIELD RAISE SW
EPS125	DG B LOCAL FIELD LOWER SW

LOCAL OPERATOR ACTION (LOA) LISTING

EPS126	DG B LOCAL ENG SHTDN RESET SW
EPS127	DG B LOCAL GEN SHTDN RESET SW
EPS128	DG B LOCAL EXCITER SHTDN SW
EPS129	DG B LOCAL EXCITER SHTDN RESET SW
EPS130	DG B LOCAL MANUAL FIELD FLASH SW
EPS131	CLOSE DG B BKR TO BUS DA (LOCAL)
EPS132	TRIP DG B BKR TO BUS DA (LOCAL)
EPS133	RESET MAIN GEN UNDERFREQ TRIP
EPS134	DELETED (NO LONGER AVAILABLE)
EPS135	LOCAL BKR 1C24D CONTROL
EPS136	LOAD SEQUENCER A: CONTROL POWER SWITCH
EPS137	LOAD SEQUENCER A: LIGHT POWER SWITCH
EPS138	LOAD SEQUENCER B: CONTROL POWER SWITCH
EPS139	LOAD SEQUENCER B: LIGHT POWER SWITCH
EPS140	MAIN GEN BKR - MANUAL TRIP
EPS141	GEN FIELD BKR - MANUAL TRIP
EPS142	GEN EXCITER BKR - MANUAL TRIP
EPS143	DG A BKR - MANUAL TRIP
EPS144	DG B BKR - MANUAL TRIP
EPS145	LOCAL BUS A5 CIRCUIT BREAKER
EPS146	LOCAL BUS C5 CIRCUIT BREAKER
EPS147	RELAY 86T1M - MAIN XFMR LOCKOUT
EPS148	RELAY 86T2-1 - UNIT AUX XFMR LOCKOUT
EPS149	RELAY 86C1M - OVERALL BACKUP LOCKOUT
EPS150	RELAY 86G - GENERATOR DIFFERENTIAL PROTECTION
EPS151	RELAY 86GC - GEN & MAIN XFMR BACKUP & FIELD FAIL
EPS152	RELAY 86T3 - EMER AUX XFMR PROTECTION
EPS153	RELAY 86T31 - EMER AUX XFMR PROTECTION
EPS154	RELAY 86T32 - EMER AUX XFMR PROTECTION
EPS155	RELAY 86T1A1-2 - 480V SWGR BUS 1A1-1A2 DIFFERENTIAL
EPS156	RELAY 86T1A3-4-5 - 480V SWGR BUS 1A3-1A4-1A5 DIFFERENTIAL
EPS157	RELAY 86T1B1-2 - 480V SWGR BUS 1B1-1B2 DIFFERENTIAL
EPS158	RELAY 86T1B3-4 - 480V SWGR BUS 1B3-1B4 DIFFERENTIAL
EPS159	RELAY 86T1C1-2 - 480V SWGR BUS 1C1-1C2 DIFFERENTIAL

LOCAL OPERATOR _TION (LOA) LISTING

EPS160	RELAY 86T1C3-4-5 - 480V SWGR BUS 1C3-1C4-1C5 DIFFERENTIAL
EPS161	RELAY 86T1DA1-2 - 480V SWGR BUS 1DA1-1DA2 DIFFERENTIAL
EPS162	RELAY 86T1DB1-2 - 480V SWGR BUS 1DB1-1DB2 DIFFERENTIAL
EPS163	RELAY 51BX-1A - 480V SWGR BUS 1A OVERCURRENT
EPS164	RELAY 51BX-1B - 480V SWGR BUS 1B OVERCURRENT
EPS165	RELAY 51BX-1C - 480V SWGR BUS 1C OVERCURRENT
EPS166	RELAY 51BX-1DA - 480V SWGR BUS 1DA OVERCURRENT
EPS167	RELAY 51BX-1DB - 480V SWGR BUS 1DB OVERCURRENT
EPS168	115 KV CIRCUIT SWITCHER #1 RACK IN/OUT
EPS169	115 KV CIRCUIT SWITCHER #2 RACK IN/OUT
EPS170	XTF4-XTF5 TIE BKR RACK IN/OUT
EPS171	APN-1FC1 POWER SELECTOR SWITCH
EPS172	XPN-7003 DISCONNECT
EPS173	XPN-7005 DISCONNECT
EPS174	XPN-7006 DISCONNECT
EPS175	XPN-7007 DISCONNECT
EPS176	XPN-7008 DISCONNECT
EPS177	XPN-6001 DISCONNECT
EPS178	XPN-6002 DISCONNECT
EPS179	XPN-6003 DISCONNECT
EPS180	XPN-6004 DISCONNECT
EPS181	XPN-6005 DISCONNECT
EPS182	XPN-6006 DISCONNECT
FWM1	VALVE BDH-547 FOR S/G BLOWDOWN TO CIRC WATER
FWM2	S/G A BLOWDOWN HX ISOL
FWM3	S/G B BLOWDOWN HX ISOL
FWM4	S/G C BLOWDOWN HX ISOL
FWM5	S/G A BLOWDOWN TO CIRC WATER ISOL
FWM6	S/G A BLOWDOWN TO NUC BLOWDOWN ISOL
FWM7	S/G B BLOWDOWN TO CIRC WATER ISOL
FWM8	S/G B BLOWDOWN TO NUC BLOWDOWN ISOL
FWM9	S/G C BLOWDOWN TO CIRC WATER ISOL
FWM10	S/G C BLOWDOWN TO NUC BLOWDOWN ISOL
FWM11	S/G A TO NUC SAMPLING ISOL

LOCAL OPERATOR ACTION (LOA) LISTING

FWM12	S/G A TO NUC SAMPLING ISOL
FWM13	S/G B TO NUC SAMPLING ISOL
FWM14	S/G B TO NUC SAMPLING ISOL
FWM15	S/G C TO NUC SAMPLING ISOL
FWM16	S/G C TO NUC SAMPLING ISOL
FWM17	S/G BLOWDOWN TO WASTE PROCESS ISOL
FWM18	S/G FPT A MANUAL LOW VACUUM RESET
FWM19	S/G FPT B MANUAL LOW VACUUM RESET
FWM20	S/G FPT C MANUAL LOW VACUUM RESET
FWM21	FWBP A WARMUP FLOW ISOL. VLV 1638A
FWM22	FWBP B WARMUP FLOW ISOL. VLV 1638B
FWM23	FWBP C WARMUP FLOW ISOL. VLV 1638C
FWM24	FWBP D WARMUP FLOW ISOL. VLV 1638D
FWM25	MS SPARGING STEAM TO DEAERATOR ISOL. VLV. 1305
FWM26	AUX STM SPARGING STM TO DA ISOL VLV AS-124
FWM27	MS SPARGING TO DEAERATOR CONTROLLER SETPOINT
FWM28	AUX STEAM SPARGING TO DEAERATOR CONTROLLER SETPOINT
FWM29	HP HEATER TRAIN A INLET ISOLATION V ALVE FW-1608A
FWM30	HP HEATER TRAIN B INLET ISOLATION VALVE FW-1608B
FWM31	HP HEATER BYPASS LINE ISOLATION VALVE FW-1609
FWM32	EXTRA BLOWDOWN FLOW FOR S/G A
FWM33	EXTRA BLOWDOWN FLOW FOR S/G B
FWM34	EXTRA BLOWDOWN FLOW FOR S/G C
FWM35	DE-ENERGIZE FORWARD FLUSH VALVES(1689 A,B,C)
FWM36	DEMIN WTR TO CST (VLV RCOL3094) MAN OVERRIDE
FWM37	DEMIN WTR TO CST (VLV RCOL3094) SET POSITION
FWM38	VALVE FV-3120 UPSTREAM ISOL VALVE (629)
FWM39	VALVE FV-3120 DOWNSTREAM ISOL VLV (630)
FWM40	FW ISOL LOOP A (1611A & 3321) TR A JUMPER
FWM41	FW ISOL LOOP B (1611B & 3331) TR A JUMPER
FWM42	FW ISOL LOOP C (1611C & 3341) TR A JUMPER
FWM43	FW CNTL BYP VLV'S (3321,3331,3341) TR A JUMPER
FWM44	FW ISOL VLV'S TRAIN B FUSES
FWM45	FW PUMP SEAL WATER TCV BYPASS VALVE FW-1660

LOCAL OPERATOR ACTION (LOA) LISTING

FWM46	EFW PUMP A BKR - MANUAL TRIP
FWM47	EFW PUMP B BKR - MANUAL TRIP
FWM48	FWB PUMP A BKR - MANUAL TRIP
FWM49	FWB PUMP B BKR - MANUAL TRIP
FWM50	FWB PUMP C BKR - MANUAL TRIP
FWM51	FWB PUMP D BKR - MANUAL TRIP
MSS1	VENT STEAM FLOW - LINE A
MSS2	VENT STEAM FLOW - LINE B
MSS3	VENT STEAM FLOW - LINE C
MSS4	ATMOS STEAM DUMP A ISOL VLV
MSS5	ATMOS STEAM DUMP B ISOL VLV
MSS6	ATMOS STEAM DUMP C ISOL VLV
MSS7	MAIN STEAM A DUMP TO COND B ISOL
MSS8	MAIN STEAM C DUMP TO COND B ISOL
MSS9	MAIN STEAM B DUMP TO COND A ISOL
MSS10	MAIN STEAM D DUMP TO COND A ISOL
MSS11	ATMOS STEAM DUMP VLV (2006) BANK 3
MSS12	ATMOS STEAM DUMP VLV (2016) BANK 3
MSS13	ATMOS STEAM DUMP VLV (2026) BANK 3
MSS14	MAIN STEAM A DUMP TO COND B (2096) BANK 1
MSS15	MAIN STEAM A DUMP TO COND B (2097) BANK 2
MSS16	MAIN STEAM C DUMP TO COND B (2106) BANK 2
MSS17	MAIN STEAM C DUMP TO COND B (2107) BANK 2
MSS18	MAIN STEAM B DUMP TO COND A (2116) BANK 1
MSS19	MAIN STEAM B DUMP TO COND A (2117) BANK 2
MSS20	MAIN STEAM D DUMP TO COND A (2126) BANK 2
MSS21	MAIN STEAM D DUMP TO COND A (2127) BANK 2
MSS22	MAIN STEAM LINE A ISOL VLV
MSS23	MAIN STEAM LINE B ISOL VLV
MSS24	MAIN STEAM LINE C ISOL VLV
MSS25	MOISTURE SEPARATOR REHEATER BYPASS VLV 2807
MSS26	EMERGENCY FW TDFP MN STM THROTTLE VLV 2865 RESET
MSS27	REHEATER AUTO/MANUAL CONTROL STATION STATUS
MSS28	REHEATER A PANEL LOADER AUTO/MANUAL STATUS

LOCAL OPERATOR ACTION (LOA) LISTING

MSS29	REHEATER B PANEL LOADER AUTO/MANUAL STATUS
MSS30	REHEATER A PANEL LOADER MANUAL OUTPUT
MSS31	REHEATER B PANEL LOADER MANUAL OUTPUT
MSS32	TDEFP STM SUP VLV 2802A BKR
MSS33	TDEFP STM SUP VLV 2802B BKR
MSS34	SG PORV 2000 AUTO/MANUAL STATUS
MSS35	SG PORV 2010 AUTO/MANUAL STATUS
MSS36	SG PORV 2020 AUTO/MANUAL STATUS
MSS37	SG PORV 2000 MANUAL POSITION
MSS38	SG PORV 2010 MANUAL POSITION
MSS39	SG PORV 2020 MANUAL POSITION
MSS40	CLOSE EXTR STEAM TO DA DRAIN VALVES
MSS41	STM DMP VLV 2006 MANUAL OPERATION
MSS42	STM DMP VLV 2016 MANUAL OPERATION
MSS43	STM DMP VLV 2026 MANUAL OPERATION
MSS44	STM DMP VLV 2096 MANUAL OPERATION
MSS45	STM DMP VLV 2097 MANUAL OPERATION
MSS46	STM DMP VLV 2106 MANUAL OPERATION
MSS47	STM DMP VLV 2107 MANUAL OPERATION
MSS48	STM DMP VLV 2116 MANUAL OPERATION
MSS49	STM DMP VLV 2117 MANUAL OPERATION
MSS50	STM DMP VLV 2126 MANUAL OPERATION
MSS51	STM DMP VLV 2127 MANUAL OPERATION
MSS52	S/G A PORV ISOL VALVE
MSS53	S/G B PORV ISOL VALVE
MSS54	S/G C PORV ISOL VALVE
NIS1	SR REACTOR TRIP CH 31
NIS2	SR REACTOR TRIP CH 32
NIS3	IR REACTOR TRIP CH 35
NIS4	IR REACTOR TRIP CH 36
NIS5	PR REACTOR TRIP LOW STPT CH 41
NIS6	PR REACTOR TRIP LOW STPT CH 42
NIS7	PR REACTOR TRIP LOW STPT CH 43
NIS8	PR REACTOR TRIP LOW STPT CH 44

LOCAL OPERATOR ACTION (LOA) LISTING

NIS9	PR REACTOR TRIP HIGH STPT CH 41
NIS10	PR REACTOR TRIP HIGH STPT CH 42
NIS11	PR REACTOR TRIP HIGH STPT CH 43
NIS12	PR REACTOR TRIP HIGH STPT CH 44
NIS13	PR REACTOR TRIP RATE NEGATIVE CH 41
NIS14	PR REACTOR TRIP RATE POSITIVE CH 41
NIS15	PR REACTOR TRIP RATE NEGATIVE CH 42
NIS16	PR REACTOR TRIP RATE POSITIVE CH 42
NIS17	PR REACTOR TRIP RATE NEGATIVE CH 43
NIS18	PR REACTOR TRIP RATE POSITIVE CH 43
NIS19	PR REACTOR TRIP RATE NEGATIVE CH 44
NIS20	PR REACTOR TRIP RATE POSITIVE CH 44
NIS21	NIS IR P-6 CH 35
NIS22	NIS IR P-6 CH 36
NIS23	NIS PR P-10 CH 41
NIS24	NIS PR P-8 CH 41
NIS25	NIS PR P-9 CH 41
NIS26	NIS PR P-10 CH 42
NIS27	NIS PR P-8 CH 42
NIS28	NIS PR P-9 CH 42
NIS29	NIS PR P-10 CH 43
NIS30	NIS PR P-8 CH 43
NIS31	NIS PR P-9 CH 43
NIS32	NIS PR P10 CH 44
NIS33	NIS PR P-8 CH 44
NIS34	NIS PR P-9 CH 44
NIS35	INCORE DET A CG/EMER/STORAGE TOP LIMIT
NIS36	INCORE DET A CG/EMER/STORAGE BOTTOM LIMIT
NIS37	INCORE DET A CALIBRATE TOP LIMIT
NIS38	INCORE DET A CALIBRATE BOTTOM LIMIT
NIS39	INCORE DET B CG/EMER/STORAGE TOP LIMIT
NIS40	INCORE DET B CG/EMER/STORAGE BOTTOM LIMIT
NIS41	INCORE DET B CALIBRATE TOP LIMIT
NIS42	INCORE DET B CALIBRATE BOTTOM LIMIT

LOCAL OPERATOR ACTION (LOA) LISTING

NIS43	INCORE DET C CG/EMER/STORAGE TOP LIMIT
NIS44	INCORE DET C CG/EMER/STORAGE BOTTOM LIMIT
NIS45	INCORE DET C CALIBRATE TOP LIMIT
NIS46	INCORE DET C CALIBRATE BOTTOM LIMIT
NIS47	INCORE DET D CG/EMER/STORAGE TOP LIMIT
NIS48	INCORE DET D CG/EMER/STORAGE BOTTOM LIMIT
NIS49	INCORE DET D CALIBRATE TOP LIMIT
NIS50	INCORE DET D CALIBRATE BOTTOM LIMIT
NIS51	INCORE DET E CG/EMER/STORAGE TOP LIMIT
NIS52	INCORE DET E CG/EMER/STORAGE BOTTOM LIMIT
NIS53	INCORE DET E CALIBRATE TOP LIMIT
NIS54	INCORE DET E CALIBRATE BOTTOM LIMIT
NIS55	SR-31 HI FLUX AT SHUTDOWN ALARM SETPOINT
NIS56	SR-32 HI FLUX AT SHUTDOWN ALARM SETPOINT
NIS57	IR-35 TEST SWITCH S3
NIS58	IR-36 TEST SWITCH S3
PCS1	POWER MISMATCH DEFEAT SW
PCS2	SOLID STATE PROT. SYS TRAIN A MODE SELECTOR SWITCH
PCS3	SOLID STATE PROT. SYS TRAIN B MODE SELECTOR SWITCH
PCS4	RCS LOW FLOW LOOP 1 CH 1
PCS5	RCS LOW FLOW LOOP 1 CH 2
PCS6	RCS LOW FLOW LOOP 1 CH 3
PCS7	RCS LOW FLOW LOOP 2 CH 1
PCS8	RCS LOW FLOW LOOP 2 CH 2
PCS9	RCS LOW FLOW LOOP 2 CH 3
PCS10	RCS LOW FLOW LOOP 3 CH 1
PCS11	RCS LOW FLOW LOOP 3 CH 2
PCS12	RCS LOW FLOW LOOP 3 CH 3
PCS13	RCS OVERPOWER DELTA T LOOP 1
PCS14	RCS OVERTEMP DELTA T LOOP 1
PCS15	RCS LOW TAVG LOOP 1
PCS16	RCS HIGH TAVG LOOP 1
PCS17	RCS LO-LO TAVG LOOP 1
PCS18	RCS OVERPOWER DELTA T LOOP 2

LOCAL OPERATOR ACTION (LOA) LISTING

PCS19	RCS OVERTEMP DELTA T LOOP 2
PCS20	RCS LOW TAVG LOOP 2
PCS21	RCS HIGH TAVG LOOP 2
PCS22	RCS LO-LO TAVG LOOP 2
PCS23	RCS OVERPOWER DELTA T LOOP 3
PCS24	RCS OVERTEMP DELTA T LOOP 3
PCS25	RCS LOW TAVG LOOP 3
PCS26	RCS HIGH TAVG LOOP 3
PCS27	RCS LO-LO TAVG LOOP 3
PCS28	PZR HIGH WATER LEVEL CH 1
PCS29	PZR HIGH WATER LEVEL CH 2
PCS30	PZR HIGH WATER LEVEL CH 3
PCS31	PZR HIGH PRESSURE CH 1
PCS32	P-11 CH 1
PCS33	PZR LOW PRESSURE REACTOR TRIP CH 1
PCS34	PZR LOW PRESSURE SI CH 1
PCS35	PZR HIGH PRESSURE CH 2
PCS36	P-11 CH 2
PCS37	PZR LOW PRESSURE REACTOR TRIP CH 2
PCS38	PZR LOW PRESSURE SI CH 2
PCS39	PZR HIGH PRESSURE CH 3
PCS40	P-11 CH 3
PCS41	PZR LOW PRESSURE REACTOR TRIP CH 3
PCS42	PZR LOW PRESSURE SI CH 3
PCS43	LP1 HIGH STEAM LINE FLOW CH 3
PCS44	LP1 HIGH STEAM LINE FLOW CH 4
PCS45	LP1 STM/FDW MISMATCH LOW FDW FLOW CH 3
PCS46	LP1 STM/FDW MISMATCH LOW FDW FLOW CH 4
PCS47	LP2 HIGH STEAM LINE FLOW CH 3
PCS48	LP2 HIGH STEAM LINE FLOW CH 4
PCS49	LP2 STM/FDW MISMATCH LOW FDW FLOW CH 3
PCS50	LP2 STM/FDW MISMATCH LOW FDW FLOW CH 4
PCS51	LP3 HIGH STEAM LINE FLOW CH 3
PCS52	LP3 HIGH STEAM LINE FLOW CH 4

LOCAL OPERATOR ACTION (LOA) LISTING

PCS53	LP3 STM/FDW MISMATCH LOW FDW FLOW CH 3
PCS54	LP3 STM/FDW MISMATCH LOW FDW FLOW CH 4
PCS55	S/G 1 LO-LO WTR LEV CH 1
PCS56	LOOP 1 LOW WTR LEV CH 1
PCS57	S/G 1 HI-HI LEV CH 1
PCS58	S/G 1 LO-LO-LO WTR LEV CH 1
PCS59	S/G 1 LO-LO WTR LEV CH 2
PCS60	LOOP 1 LOW WTR LEV CH 2
CS61	S/G 1 HI-HI LEV CH 2
PCS62	S/G 1 LO-LO-LO WTR LEV CH 2
PCS63	S/G 1 LO-LO WTR LEV CH 3
PCS64	S/G 1 HI-HI WTR LEV CH 3
PCS65	S/G 1 LO-LO-LO WTR LEV CH 3
PCS66	S/G 2 LO-LO WTR LEV CH 1
PCS67	LOOP 2 LOW WTR LEV CH 1
CS68	S/G 2 HI-HI LEV CH 1
PCS69	S/G 2 LO-LO-LO WTR LEV CH 1
PCS70	S/G 2 LO-LO WTR LEV CH 2
PCS71	LOOP 2 LOW WTR LEV CH 2
PCS72	S/G 2 HI-HI LEV CH 2
PCS73	S/G 2 LO-LO-LO WTR LEV CH 2
PCS74	S/G 2 LO-LO WTR LEV CH 3
PCS75	S/G 2 HI-HI LEV CH 3
PCS76	S/G 2 LO-LO-LO WTR LEV CH 3
PCS77	S/G 3 LO-LO WTR LEV CH 1
PCS78	LOOP 3 LOW WTR LEV CH 1
PCS79	S/G 3 HI-HI LEV CH 1
PCS80	S/G 3 LO-LO-LO WTR LEV CH 1
PCS81	S/G 3 LO-LO WTR LEV CH 2
PCS82	LOOP 3 LOW WTR LEV CH 2
PCS83	S/G 3 HI-HI LEV CH 2
PCS84	S/G 3 LO-LO-LO WTR LEV CH 2
PCS85	S/G 3 LO-LO WTR LEV CH 3
PCS86	S/G 3 HI-HI LEV CH 3

LOCAL OPERATOR ACTION (LOA) LISTING

PCS87	S/G 3 LO-LO LO WTR LEV CH 3
PCS88	LOOP 1 LOW STEAM LINE PRESSURE
PCS89	P1-P2 HIGH STEAM LINE DIFF PRESSURE CH 2
PCS90	P1-P2 HIGH STEAM LINE DIFF PRESSURE CH 2
PCS91	P1-P2 HIGH STEAM LINE DIFF PRESSURE CH 3
PCS92	P1-P2 HIGH STEAM LINE DIFF PRESSURE CH 3
PCS93	P1-P2 HIGH STEAM LINE DIFF PRESSURE CH 4
PCS94	P1-P2 HIGH STEAM LINE DIFF PRESSURE CH 4
PCS95	P2-P3 HIGH STEAM LINE DIFF PRESSURE CH 2
PCS96	P2-P3 HIGH STEAM LINE DIFF PRESSURE CH 2
PCS97	LOOP 2 LOW STEAM LINE PRESSURE
PCS98	P2-P3 HIGH STEAM LINE DIFF PRESSURE CH 3
PCS99	P2-P3 HIGH STEAM LINE DIFF PRESSURE CH 3
PCS100	P2-P3 HIGH STEAM LINE DIFF PRESSURE CH 4
PCS101	P2-P3 HIGH STEAM LINE DIFF PRESSURE CH 4
PCS102	P1-P3 HIGH STEAM LINE DIFF PRESSURE CH 2
PCS103	P1-P3 HIGH STEAM LINE DIFF PRESSURE CH 2
PCS104	P1-P3 HIGH STEAM LINE DIFF PRESSURE CH 3
PCS105	P1-P3 HIGH STEAM LINE DIFF PRESSURE CH 3
PCS106	LOOP 3 LOW STEAM LINE PRESSURE
PCS107	P1-P3 HIGH STEAM LINE DIFF PRESSURE CH 4
PCS108	P1-P3 HIGH STEAM LINE DIFF PRESSURE CH 4
PCS109	CNMT PRESSURE HI-3 CH 1
PCS110	CNMT PRESSURE HI-3 CH 2
PCS111	CNMT PRESSURE HI-1 CH 1
PCS112	CNMT PRESSURE HI-2 CH 1
PCS113	CNMT PRESSURE HI-3 CH 3
PCS114	CNMT PRESSURE HI-1 CH 2
PCS115	CNMT PRESSURE HI-2 CH 2
PCS116	CNMT PRESSURE HI-3 CH 4
PCS117	CNMT PRESSURE HI-1 CH 3
PCS118	CNMT PRESSURE HI-2 CH 3
PCS119	IR HIGH FLUX ROD STOP CH 35
PCS120	IR HIGH FLUX ROD STOP CH 36

LOCAL OPERATOR ACTION (LOA) LISTING

PCS121	IR HIGH FLUX ROD STOP CH 41
PCS122	IR HIGH FLUX ROD STOP CH 42
PCS123	IR HIGH FLUX ROD STOP CH 43
PCS124	IR HIGH FLUX ROD STOP CH 44
PCS125	AUTO RODS IN
PCS126	AUTO RODS OUT
PCS127	OPDT ROD STOP LOOP 1
PCS128	OPDT ROD STOP LOOP 2
PCS129	OPDT ROD STOP LOOP 3
PCS130	OTDT ROD STOP LOOP 1
PCS131	OTDT ROD STOP LOOP 2
PCS132	OTDT ROD STOP LOOP 3
PCS133	CTRL BK D WITHDRAWAL LIMIT ; ROD STOP
PCS134	BLK AUTO ROD WITHDRAWAL : C-5
PCS135	TUR IMP CH PRESS > 10% F.P. CH 3
PCS136	TUR IMP CH PRESS > 10% F.P. CH 4
PCS137	AUCT TAVG > LOOP 1 TAVG
PCS138	LOOP 1 TAVG > AUCT TAVG
PCS139	AUCT TAVG >LOOP 2 TAVG
PCS140	LOOP 2 TAVG > AUCT TAVG
PCS141	AUCT TAVG > LOOP 3 TAVG
PCS142	LOOP 3 TAVG > AUCT TAVG
PCS143	AUCT DELTA T > LOOP 1 DELTA T
PCS144	LOOP 1 DELTA T > AUCT DELTA T
PCS145	AUCT DELTA T > LOOP 2 DELTA T
PCS146	LOOP 2 DELTA T > AUCT DELTA T
PCS147	AUCT DELTA T > LOOP 3 DELTA T
PCS148	LOOP 3 DELTA T > AUCT DELTA T
PCS149	BANK A LOW LIMIT
PCS150	BANK A LO-LO LIMIT
PCS151	BANK B LOW LIMIT
PCS152	BANK B LO-LO LIMIT
PCS153	BANK C LOW LIMIT
PCS154	BANK C LO-LO LIMIT

LOCAL OPERATOR ACTION (LOA) LISTING

PCS155	BANK D LOW LIMIT
PCS156	BANK D LO-LO LIMIT
PCS157	SUDDEN LOSS OF TUR LOAD
PCS158	DEV:TREF/TAVG,HI 1,LOAD REJ
PCS159	DEV:TREF/TAVG,HI 2,LOAD REJ
PCS160	DEV:TREF/TAVG,HI 1,TURBINE TRIP
PCS161	DEV:TREF/TAVG HI 2,TURBINE TRIP
PCS162	LOW PRZR LEV, BLKS HTRS&LCV459
PCS163	LOW PRZR LEV, BLKS HTRS&LCV460
PCS164	HIGH DEV PZR LEV/BU HTRS ON
PCS165	LOW DEV,PRZR PRESS/BU HTRS ON
PCS166	LOW PZR PRESSURE
PCS167	HI PZR PRESSURE
PCS168	HI PZR PRESSURE ENABLE PORV PCV445A
PCS169	HI PZR DEVIATION
PCS170	HI PZR DEVIATION ENABLE PORV PCV444B
PCS171	LOW LEVEL DEVIATION
PCS172	HIGH PZR LEVEL
PCS173	SG1 STM/FDW MISMATCH : FDW>STM
PCS174	SG1 STM/FDW MISMATCH : STM>FDW
PCS175	SG2 STM/FDW MISMATCH : FDW>STM
PCS176	SG2 STM/FDW MISMATCH : STM>FDW
PCS177	SG3 STM/FDW MISMATCH : FDW>STM
PCS178	SG3 STM/FDW MISMATCH : STM>FDW
PCS179	SG1 LOW LEVEL DEVIATION
PCS180	SG1 HIGH LEVEL DEVIATION
PCS181	SG2 LOW LEVEL DEVIATION
PCS182	SG2 HIGH LEVEL DEVIATION
PCS183	SG3 LOW LEVEL DEVIATION
PCS184	SG3 HIGH LEVEL DEVIATION
PCS185	RX TRIP BYPASS BKR A - CONNECTED
PCS186	RX TRIP BYPASS BKR A - OPEN
PCS187	RX TRIP BYPASS BKR A - CLOSE
PCS188	RX TRIP BYPASS BKR B - CONNECTED

LOCAL OPERATOR ACTION (LOA) LISTING

PCS189	RX TRIP BYPASS BKR B - OPEN
PCS190	RX TRIP BYPASS BKR B - CLOSE
PCS191	RX TRIP BKR A - CONNECTED
PCS192	RX TRIP BKR A - OPEN
PCS193	RX TRIP BKR A - CLOSE
PCS194	RX TRIP BKR B - CONNECTED
PCS195	RX TRIP BKR B - OPEN
PCS196	RX TRIP BKR B - CLOSE
PCS197	CHAN II SG A PB-474C
PCS198	CHAN II SG B PB-484C
PCS199	CHAN II SG C PB-494C
PCS200	CHAN III SG A PB-475C
PCS201	CHAN III SG B PB-485C
PCS202	CHAN III SG C PB-495C
PCS203	CHAN IV SG A PB-476C
PCS204	CHAN IV SG B PB-486C
PCS205	CHAN IV SG C PB-496C
PCS206	CHAN III SG A FB-477A
PCS207	CHAN III SG B FB-487A
PCS208	CHAN III SG C FB-497A
PCS209	CHAN IV SG A FB-476A
PCS210	CHAN IV SG B FB-486A
PCS211	CHAN IV SG C FB-496A
PRS1	PRT N2 SUPPLY REGULATOR STATUS
PRS2	PRT N2 SUPPLY REGULATOR SETPOINT, PSIG
PRS3	PRT VENT TO CNM 8048
PRS4	PRS PCV-444D SPRAY BYPASS VALVE
PRS5	PRS PCV-444C SPRAY BYPASS VALVE
PRS6	PRS VENT LINE ISOL VALVE
PRS7	PRT VENT TO GAS ANALYZER/WPS
PRS8	AVAILABLE FOR USE
PRS9	RCDT PUMP A MANUAL CONTROL
PRS10	RCDT PUMP B MANUAL CONTROL
PRS11	RCDT DISCHARGE ISOL VLV

LOCAL OPERATOR ACTION (LOA) LISTING

PRS12	RCDT PUMP DISCH VLV TO PRT
PRS13	RCDT RECIRC VALVE
PRS14	RCDT HX BYPASS VLAVE
PRS15	RCDT HX BYPASS VLAVE
PRS16	RCDT LVL CNTRL VALVE 1003 MANUAL/AUTO SELECTION
PRS17	RCDT LVL CNTRL VALVE 1003 MANUAL POSITION SELECTION
RCS1	RCP SHAFT VIB ALERT SETPT
RCS2	RCP SHAFT VIB DANGER SETPT
RCS3	RCP FRAME VIB ALERT SETPT
RCS4	RCP FRAME VIB DANGER SETPT
RCS5	RCS LOOP 2 DRAIN ISOL VALVE
RCS6	RCP A BKR
RCS7	RCP B BKR
RCS8	RCP C BKR
RCS9	RX HEAD VENT VLV 8095A BKR
RCS10	RX HEAD VENT VLV 8095B BKR
RCS11	RX HEAD VENT VLV 8096A BKR
RCS12	RX HEAD VENT VLV 8096B BKR
RCS13	PZR PORV BLK VLV 8000A BKR
RCS14	PZR PORV BLK VLV 8000B BKR
RCS15	PZR PORV BLK VLV 8000C BKR
RCS16	CCM (TRAIN-A) P-1 INPUT SWITCH, INPUT FROM PT-402
RCS17	CCM (TRAIN-A) P-2 INPUT SWITCH, INPUT FROM PT-403
RCS18	CCM (TRAIN-A) P-3 INPUT SWITCH, INPUT FROM PT-455
RCS19	CCM (TRAIN-B) P-1 INPUT SWITCH, INPUT FROM PT-402
RCS20	CCM (TRAIN-B) P-2 INPUT SWITCH, INPUT FROM PT-403
RCS21	CCM (TRAIN-B) P-3 INPUT SWITCH, INPUT FROM PT-457
RCS22	CCM (TRAIN-A) LOOP-1 TH INPUT SW., INPUT FROM TE-413
RCS23	CCM (TRAIN-A) LOOP-1 TC INPUT SW., INPUT FROM TE-410
RCS24	CCM (TRAIN-A) LOOP-2 TH INPUT SW., INPUT FROM TE-423
RCS25	CCM (TRAIN-A) LOOP-2 TC INPUT SW., INPUT FROM TE-420
RCS26	CCM (TRAIN-B) LOOP-1 TH INPUT SW., INPUT FROM TE-423
RCS27	CCM (TRAIN-B) LOOP-1 TC INPUT SW., INPUT FROM TE-420
RCS28	CCM (TRAIN-B) LOOP-2 TH INPUT SW., INPUT FROM TE-433

LOCAL OPERATOR ACTION (LOA) LISTING

RCS29	CCM (TRAIN-B) LOOP-2 TC INPUT SW., INPUT FROM TE-430
RCS30	PZR HTR BU GRP 1 BKR - MANUAL TRIP
RCS31	PZR HTR BU GRP 2 BKR - MANUAL TRIP
RCS32	PZR HTR CNTRL GRP BKR - MANUAL TRIP
RCS33	RCP A BKR - MANUAL TRIP
RCS34	RCP B BKR - MANUAL TRIP
RCS35	RCP C BKR - MANUAL TRIP
RCS36	PZR MANWAY
RCS37	S/G HOT LEG MANWAY LOOP A
RCS38	S/G HOT LEG MANWAY LOOP B
RCS39	S/G HOT LEG MANWAY LOOP C
RCS40	S/G COLD LEG MANWAY LOOP A
RCS41	S/G COLD LEG MANWAY LOOP B
RCS42	S/G COLD LEG MANWAY LOOP C
RCS43	S/G HOT LEG NOZZLE DAM LOOP A
RCS44	S/G HOT LEG NOZZLE DAM LOOP B
RCS45	S/G HOT LEG NOZZLE DAM LOOP C
RCS46	S/G COLD LEG NOZZLE DAM LOOP A
RCS47	S/G COLD LEG NOZZLE DAM LOOP B
RCS48	S/G COLD LEG NOZZLE DAM LOOP C
RCS49	RCS LOOP 1 DRAIN ISOL VALVE
RCS50	RCS LOOP 3 DRAIN ISOL VALVE
RCS51	RCS FLANGE LEAKOFF INNER VALVE
RCS52	RCS FLANGE LEAKOFF OUTER VALVE
RCS53	MID-LOOP MONITOR DISCONNECT SWITCH
RHR1	RHR TR A CVC LTDN ISO VLV
RHR2	RHR TR B CVC LTDN ISO VLV
RHR3	RHR CL INJ HDR ISO VLV
RHR4	RHR CL INJ HDR ISO VLV
RHR5	RHR PPS RTN TO RWST
RHR6	ACCUM A ISO VLV 8808A BKR
RHR7	ACCUM B ISO VLV 8808B BKR
RHR8	ACCUM C [illegible] VLV 8808C BKR
RHR9	RHR PP [illegible] INLET VLV 8701A BKR

LOCAL OPERATOR ACTION (LOA) LISTING

RHR10	RHR PP B INLET VLV 8701B BKR
RHR11	RHR PP A INLET VLV 8702A BKR
RHR12	RHR PP B INLET VLV 8702B BKR
RHR13	RHR PP A TO CHRG PP 8706A BKR
RHR14	RHR PP B TO CHRG PP 8706B BKR
RHR15	RHR VALVE 602A BREAKER POSITION
RHR16	RHR VALVE 602B BREAKER POSITION
RHR17	RHR 602A VALVE POSITION
RHR18	RHR 602B VALVE POSITION
RHR19	RHR PUMP A BKR - MANUAL TRIP
RHR20	RHR PUMP B BKR - MANUAL TRIP
RHR21	RHR SUCT RELIEF 8708A LEAK POSITION
RHR22	RHR SUCT RELIEF 8708B LEAK POSITION
RHR23	RHR PUMP A VENT
RHR24	RHR PUMP B VENT
RMS1	RAD LEVEL IN RCS (UC/CC)
RMS2	RAD LEVEL IN FUEL HANDLING BLDG
RMS3	RAD LEVEL IN WASTE GAS VENT
RMS4	RAD LEVEL IN AUX BLDG
RMS5	RAD LEVEL IN CONTROL RM AIR SUPPLY
RMS6	RM-G6 (BRIDGE AREA) SET POINTS ADJUSTMENT
RMS7	RM-G14 (INCOR INSTR AREA) SET POINTS ADJUSTMENT
TUR1	EHC ELECTRICAL MALFUNCTION RESET
TUR2	AUXILIARY BOILER SWITCH
TUR3	105VAC HOUSE PWR TO EHC CABINET
TUR4	EHC P.M.G. TEST
TUR5	DELETED (NO LONGER AVAILABLE)
TUR6	AUX BOILER TRANSFER PUMP A
TUR7	AUX BOILER TRANSFER PUMP B
TUR8	AUX BOILER FEEDWATER PUMP A
TUR9	AUX BOILER FEEDWATER PUMP B
TUR10	H2 SEAL OIL PCV BYPASS VLV
TUR11	TURBINE TRIP AT FRONT STANDARD
TUR12	GEN CASING VENT VALVE

LOCAL OPERATOR ACTION (LOA) LISTING

TUR13	GEN MACHINE GAS PRESSURE REGULATOR SETPOINT
TUR14	HYDROGEN SUPPLY SECTION (1,2,3)
TUR15	GEN SCW PUMP A SWITCH POSITION
TUR16	GEN SCW PUMP B SWITCH POSITION
TUR17	GEN RECIRC SEAL OIL PUMP SWITCH POSITION
TUR18	GEN SEAL OIL VACUUM PUMP SWITCH POSITION
TUR19	GEN MAIN SEAL OIL PUMP SWITCH POSITION
TUR20	GEN EMERGENCY SEAL OIL PUMP SWITCH POSITION
TUR21	TURBINE SUPERVISORY INSTRUMENTATION TRIP DISABLE SWITCH
TUR22	EBOP POWER SUPPLY BREAKER AT DPN2X
TUR23	125VDC CNTRL PWR TO EHC CABINET
TUR24	TURBINE FAST COOLDOWN MULTIPLIER
TUR25	TURBINE FAST WARMUP MULTIPLIER
TUR26	PMG MALFUNCTION LIGHT
TUR27	EHC HYD PUMP DISCH BYP VLV 20877

SIMULATOR OPERATING LIMITS

1. RHR PUMPS OPERATING AT SHUTOFF HEAD WITH NO FLOW
2. RHR TEMP EXCEEDED SATURATION TEMP
3. RHR PRESSURE EXCEEDED 900 PSIG
4. CS PUMPS OPERATING AT SHUTOFF HEAD WITH NO FLOW
5. CCW PRESSURE EXCEEDED 225 PSIG
6. VCT TEMP EXCEEDED SATURATION TEMP
7. CVCS DEMINERALIZER TEMPERATURE LESS THAN 60 DEGF
8. BA STORAGE TANK TEMPERATURE LESS THAN 60 DEGF
9. #1 SEAL WATER INLET TEMP EXCEEDED 245 DEGF
10. #1 SEAL LEAKAGE RATE <0.2 GPM AND PUMP ON
11. FUEL TEMPERATURE EXCEEDED 4700 DEGF
12. CLAD TEMPERATURE EXCEEDED 2250 DEGF
13. CORE VOIDING > 10% WITH RODS NOT ON BOTTOM
14. HYDROGEN GENERATION EXCEEDED 1000 LBM
15. CORE T/C TEMPERATURE EXCEEDED 2300 DEGF
16. A LUBE OIL OR EHC FLUID TEMP > 150 DEGF
17. A MAIN TURBINE BEARING EXCEEDED 15 MILS VIBRATION
18. HEAT-UP RATE OF TURBINE SHELL > 85 DEGF/HR
19. HEAT-UP RATE OF TURBINE STEAM CHEST > 150 DEGF/HR
20. RCDT TEMP EXCEEDED SATURATION TEMP
21. SC WATER TEMP EXCEEDED SATURATION TEMP
22. A 125 VDC BATTERY VOLTAGE DECREASED TO < 50 VOLTS
23. RB PRESSURE LESS THAN 10 PSIA
24. RB PRESSURE GREATER THAN 60 PSIG
25. MAIN FEED PUMP DISCHARGE PRESS > 1720 PSIG
26. CW TEMP EXCEEDED SATURATION TEMP
27. NAOH TANK LEVEL DROPPED TO ZERO PERCENT
28. RCS SYSTEM PRESSURE EXCEEDED 3200 PSIA
29. RCS SYSTEM PRESSURE LESS THAN 2 PSIA
30. RCS TEMPERATURE EXCEEDED 650 DEGF
31. RCS PRESS/TEMP OUTSIDE OPERATING BOUNDS
32. RCS COOLDOWN RATE EXCEEDED 100 DEGF/HR
33. PZR - SPRAY TEMP DELTA T EXCEEDED
34. RCP BEARING TEMPERATURE EXCEEDED 600 DEGF
35. SFP LEVEL DROPPED BELOW TOP OF STORED FUEL (437')
36. SFP WATER TEMP EXCEEDED 212 DEGF
37. TDEFP DISCHARGE PRESSURE EXCEEDED 1500 PSIG

1.4 Operating Procedures for Virgil C. Summer Nuclear Station (VCSNS)

The operating procedures used on the simulator are a controlled set of VCSNS operating procedures. There are no differences between the simulator procedures and the plant procedures.

The VCSNS curve book is also used on the simulator with the difference being that the reactor core figures are from the previous core cycle. The plant's curve book is amended prior to the first criticality of each core cycle; the simulator's curve book is updated when new data defining the latest core is loaded into the simulator. The curve book is a controlled document.

2.0 DESIGN DATA

2.1 List of Data Utilized to Support Construction of Simulator and Define Virgil C. Summer Nuclear Station

The simulator design data base is maintained from Virgil C. Summer Nuclear Station data. All of the reference plant documents used in the simulator design data base are located at the Nuclear Training Center or at the Virgil C. Summer Nuclear Station.

2.2 Description of Configuration Management System

A Simulator Configuration Management System (CMS) is used to maintain a record of the reference plant configuration and provide the capability to identify the impact of changes in the reference plant on the simulator. A description of the Configuration Management System is provided in Appendix 1 along with a list of reference plant documents (Table 1).

2.3 Simulation Model Descriptions

The simulation models are described in the Virgil C. Summer Simulator Design Basis Document Books (SDBD) which are available on site at the Virgil C. Summer Nuclear Station Nuclear Training Center. Appendix 2 to this section lists the Table of Contents for the all the SDBDs. Appendix 3 to this section provides a Table of Contents from a representative SDBD (RCS-10.2.5).

2.4 Data Referenced to Support Verification Testing

2.4.1 General

Section 1 of IST-1.0 (Appendix 1 to Section 3.0.1 of this submittal) provides the testing criteria used for Certification Testing. Listed in Sections 2.4.2 and 2.4.3 are documents referred to while verifying that the simulator's performance mimics that of the plant.

In addition, a Simulator Test Certification Panel was formed to assist in preparation of malfunction and transient tests and to assess the test results. The recommendations of the test panel were relied upon for all transients and malfunctions where actual plant results or plant design data were not available.

The following is a list of documents referred to while verifying that the simulator's performance mimics that of the plant.

2.4.2 Steady State Tests

- 100% Power - IPCS Printout, June 06, 1990
- 100% Power - Virgil C. Summer 100% Power Data - Letter WVCS-90-32 Dated 11/15/90 Khwaja to Warner
- 75% Power - IPCS Printout, May 29, 1990
- 50% Power - IPCS Printout, May 27, 1990

2.4.3 Transient and Normal Operations Test

- LER Test: Reactor Trip Following Loss of Inverter
  XIT-5904
  LER 87-0015 Trip Package
- LER Test: Reactor Trip After Loss of Power to I&C
  Process Control Cabinet XPN-7008
  ONO 87-0027 Trip Package
- LER Test: Safety Injection On Test Closure Of MSIV
  ONO 88-0027 Trip Package
- LER Test: Reactor Trip Due to High Positive Flux Rate
  LER 85-0003 Trip Package
- LER Test: Safety Injection Activation Due to Pressurizer Spray Valve Failure
  LER 85-0034 Trip Package

- Virgil C. Summer Operating Procedures
- Virgil C. Summer Final Safety Analysis Report

2.4.4 Simulator Test Certification Panel

ANS-3.5 requires test abstracts of selected malfunctions and transients to be submitted, comparing the simulator results to actual plant results, analytical or design data, or to results from similar plants.

ANS-3.5, Appendix A, Section A.3.3 (2) further states, "Compare transient tests for which no design data or actual plant response is available to best estimates. FSAR transients are based on 'worst case' situations and as such may be inappropriate for real time dynamic simulation comparison."

In addressing the "best estimate" cases, the NRC requires that documentation justifying the bases for the "best estimate" be submitted with the certification package. Mr. Neal K. Hunnemuller (USNRC-Operations Licensing Branch) released a document entitled "Simulation Facility Evaluation Program" which further clarified the NRC's position on adequate documentation. Mr. Hunnemuller stated, in part, that "If the baseline data was the judgment of a panel of experts (e.g., not actual plant results of design data), then documentation of their review, sufficient for a third party to evaluate the adequacy of the test(s) results, should be included. This documentation may include such items as the makeup and qualification of the panel and any differing professional opinions as to the outcome of the test(s)."

To meet these requirements, a Simulator Test Certification Panel was formed, and its purpose, member qualifications, and method of operation are listed in Appendix 4. The panel was convened many times to review and approve the applicable test procedures, the test results and procedure revisions recommended during testing. Minutes of these meetings are on file.

CONFIGURATION MANAGEMENT SYSTEM (CMS)

1. The CMS is used in the maintenance and modification of the Virgil C. Summer Simulator. This system is designed to run on a stand alone SUN workstation and provides an interface with certain initialization files that are maintained on the simulator computer.

2. The CMS resides on a SUN SPARC workstation and utilizes SUN Unify, a relational database management system. The system provides South Carolina Electric & Gas Company with the capability to identify the current configuration replicated by the simulator, the impact of changes on the reference plant to the simulator and its documentation, and to track the implementation of changes to the simulator and its documentation caused by the reference plant changes.

3. The CMS database contains discrete record types for reference plant documents, various types of simulated components, and simulator documents; and includes explicit relationships between the reference plant document records and the other simulator-related records. From this large database are generated many report types for effective simulator configuration management. These report types include data files for initializing component handler data structures on the simulator; data sheets for the documentation of simulated components for system models; and listings of simulated components, simulator documents, and simulator model programs which are affected by particular plant reference documents. These database outputs, along with ancillary data structures and reports for tracking simulator modifications resulting from plant changes, provide South Carolina Electric & Gas Company with the tools necessary for effective simulator configuration management.

4. The CMS database design includes records for all simulator plant components. The record types include Local Operator Actions, pumps, valves, heat exchangers, breakers, bistables, controllers, transmitters, meters, recorders, status lights, integrated plant computer points, and others. Fields include not only plant ID names, descriptions and assigned simulator system designations,

but also the data used by the engineers to model the component or feature, and a reference plant document from which the data was obtained. Record types exist in the database from all reference plant documents used in the manufacture of the simulator. These reference plant documents are actually related to the component data base records whenever data from the document is listed in a component record field.

5. One function provided by the CMS database is the ability to initialize important component handler data structures on the simulator from the SUN workstation. Data from simulated component's records are extracted from the database, formatted appropriately, and transmitted to the simulator using a simple command interface. This initialization strategy provides a mechanism for ensuring synchronism between data points in the CMS database and simulator.

6. Another function provided by the CMS, is the tracking of Simulator Discrepancy Reports (SDRs). SDRs are the controlling documents for making changes to the simulator. SDRs are written when plant changes require making modifications to the simulator or when problems are identified by simulator users. A unique ID is assigned to each SDR. Fields in each SDR record include title, date originated, due date, priority, description of problem, and actions to duplicate the problem. Other fields provide extensive cross referencing capability for tracking purposes, so that SDRs can be grouped by system, type, or class and related to required test procedures.

7. The major functions of the CMS include:

   a. Providing a record of all reference plant data available for use in the design and manufacture of the simulator. This record identifies the data which was utilized, where it was used and includes the purpose of the usage.

   b. Providing a reference record for each system and component of the plant, as identified in the database.

   c. Providing test records as separate stand-alone documents.

d. Maintaining in a separate database, records of test, and inspections of the simulator hardware and software to verify compliance to requirements, to document results of test and inspections, and to document corrective actions.

e. Indexes data entries as applicable to systems, models, modules, and components.

8. Reports provide the capability for tracking reference plant modifications and their effects on the simulator. One data structure would allow relationships to be established between a plant modification number and resulting revisions to plant documents. From this information, reports are generated which detail the simulated components, system models, and simulator documentation affected by the revised plant documents associated with the plant modifications. Simulator Discrepancy Reports are then generated to affect and track simulator modifications resulting from plant modifications.

9. The Configuration Management System (CMS) is a data management tool which also interacts with active simulation models to specify component handler characteristics.

   The data management part allows for tracking of documents used to specify how the simulation models should be configured, allows for inclusion of test data, and also provides a mechanism for tracking design changes or observed simulator discrepancies.

   The interactive part of CMS is composed of component data records whose fields require inputs for operating characteristics and the references from which they were obtained. These records are standardized within component types (valves, controllers, etc.) and require the same inputs for all simulated components in each type. The hardware in the operating plant is not as neatly standardized; there are large, small, safety related, insignificant, and diversely powered components within each classification, with equally diverse degrees of documentation. Most of the data has been gathered, checked, put into a form compatible with the data fields, and loaded into the computer.

Certification and on-site acceptance tests have been performed with the CMS data in place, and an evaluation of the completeness of the data has been made.

The simulation must accurately mimic the responses of the Virgil C. Summer Nuclear Station in order to properly train and evaluate the plant operating staff. The criteria for this objective are clearly defined in NUREGs 1021 and 1258. ANSI/ANS-3.5 provides the requirements for documenting this compliance to these criteria. Second, the computer models used to simulate core physics, fluid flow and thermodynamic heat transfer response to operator manipulations of control devices mounted on the simulator control board panels are the heart of the simulator and are essentially unaffected by CMS inputs. The complex relationships between power, temperature, pressure, flow, etc., are fixed in nodal equations and their coefficient calculations. The specific plant data references for these derivations are provided in the model's documentation. The CMS inputs specify the response of components to the parameters generated by the model codes. While still very observable, these effects are relatively less important to the overall simulation than the responses of the model codes.

The simulator response during the recent acceptance testing has been satisfactory. Simulator Deficiency Reports have been written to correct minor problems.

The recent modifications have added features and increased operational fidelity, and the present simulation conforms to the testing criteria referred to herein.

With respect to the CMS data evaluation, there are several areas where the CMS data is incomplete and are discussed below:

First, there are many references in the data records which show "NOT FOUND." These result from references for simulated components not being available at the time of data entry; they do not affect simulation.

Second, for many components, plant data does not exist to complete all of the data fields in the CMS record, and assumptions were made for those points. In most cases, these assumptions were provided by Virgil C. Summer Nuclear Station engineering staff.

Third, CMS data field references for power loss effects on mechanical components are meaningless, but do show up as data voids.

Fourth, there are a limited number of cases where actual data was available, loaded into the CMS, and had an adverse effect on simulation. Whenever this was encountered, a SDR was written defining the discrepancy and the evaluation criteria, the previously used data reinserted into the CMS field, the simulator response retested, the SDR closed and its number placed in CMS as a reference for that data. There are a number of possible reasons for this, but the primary concern is to maintain simulator fidelity to the plant. These cases have been evaluated before using this justification method and are few enough to be controllable. They will be re-evaluated whenever a related plant change is made.

In summary, the CMS is a dynamic, evolving data base that has recently been installed in the Virgil C. Summer Simulator. Training fidelity has been improved, is maintained by testing, and is acceptable for operator testing and training. As the CMS matures with use, the data voids and differences will decrease in number.

TABLE 1

SOURCE DOCUMENTS
USED IN
CONFIGURATION MANAGEMENT SYSTEM

DOC NUM	DOC NAME	REV	DATE	TITLE
1900	10319-160086-001	0	12/18/90	SIMULATOR SOFTWARE UPGRADE DATA VALUE JUSTIFICATION
1899	10319-160086-002	0	12/18/90	DATA EXCEPTION - USE CURRENT VALUE PENDING VERIFICATION
1729	10319-160086-003	0	01/24/91	POWER JUSTIFICATION FOR MECHANICAL DEVICES
3	108D837	0	10/16/84	FUNCTIONAL DIAGRAMS (1MS-41-011)
4	108D932	0	02/02/84	PROCESS CONTROL BLOCK DIAGRAM (1MS-51-032)
1660	1MS-09-175-1	0	07/19/77	REACTOR MAKEUP WATER PUMP
1665	1MS-14-045	0	11/27/89	SERVICE WATER BOOSTER PUMP
617	1MS-17-042	0	11/15/72	DRIVE STEAM TURBINE CONTROL DIAGRAM
1891	1MS-17-110-2	0	10/20/76	FW BOOSTER PUMP MOTOR DATA SHEET
1892	1MS-17-114	0	07/30/76	FW BOOSTER PMP CURVE
623	1MS-17-125	0	12/23/70	TERRY TURBINE LUBE SCHEMATIC
1890	1MS-17-146-1	0	04/13/77	EMERG FW PUMP MOTOR DATA SHEET
1662	1MS-17-150	0	08/19/77	FEED PUMP CURVE
261	1MS-19-085	0	05/06/75	VAC PUMP
592	1MS-19-098	0	05/19/75	CW PUMP CURVE
1661	1MS-19-163	0	03/24/76	OPEN CYCLE BOOSTER PUMP CURVE
1181	1MS-20-248-1	0	10/20/75	INDUSTRIAL COOLER PUMP
1419	1MS-20-291	0	06/01/75	INCORE INSTR SUMP PUMP
1635	1MS-25-007	0	01/14/74	3/4" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1624	1MS-25-009	0	01/14/74	2" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1625	1MS-25-011	0	01/14/74	3" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1628	1MS-25-012	0	10/07/81	3" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1610	1MS-25-013	0	04/16/74	4" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1591	1MS-25-015	0	03/21/72	3/8" AIR OPERATED CONTROL VALVE DWG

DOC NUM	DOC NAME	REV	DATE	TITLE
1611	1MS-25-016	0	03/21/72	3/8" AIR OPERATED CONTROL VALVE DWG
1606	1MS-25-034	0	03/05/71	3/4" 1500# AIR OPERATED CONTROL VALVE ASSY
820	1MS-25-038	0	03/05/71	3/4" 1500# AIR OPERATED CONTROL VALVE ASSY
1630	1MS-25-041	0	03/05/71	1" 1500# AIR OPERATED CONTROL VALVE ASSY
1634	1MS-25-042	0	03/05/71	1" 1500# AIR OPERATED CONTROL VALVE ASSY
1605	1MS-25-043	0	03/05/71	1" 1500# AIR OPERATED CONTROL VALVE ASSY
1588	1MS-25-053	0	02/28/71	3/4" 1500# AIR OPERATED CONTROL VALVE ASSY
1060	1MS-25-151	0	04/28/87	2" NUCLEAR DIAPH VALVE AIR OPERATED HI PRESSURE
1612	1MS-25-167	0	01/15/74	1" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1627	1MS-25-170	0	01/15/74	3/4" NUCLEAR DIAPH VALVE AIR OPERATED HIGH PRESSURE
1595	1MS-25-596	0	10/11/76	3" - 40 DIAPH AGM VALVE AIR OPERATED
1594	1MS-25-597	0	11/05/76	4" - 40 DIAPH AGM VALVE AIR OPERATED
1616	1MS-25-710	0	03/14/78	6" - 150# WELD END GATE VALVE
1617	1MS-25-755	0	08/14/78	6" - 150# WELD END GATE VALVE
624	1MS-27-043	0	12/10/82	TURBINE CONTROL DIAGRAM
619	1MS-27-061	0	10/11/82	STATOR WINDING CW SYSTEM
620	1MS-27-063	0	04/28/72	SEAL OIL UNIT
618	1MS-27-064	0	04/10/80	MAIN GENERATOR GAS CONTROL PIPING DIAGRAM
1587	1MS-50-046	0	09/27/73	3" 1500# AIR OPERATED CONTROL VALVE ASSY
1607	1MS-50-047	0	03/03/71	3" 1500# AIR OPERATED CONTROL VALVE ASSY
1589	1MS-50-157	0	09/28/73	1" 1500# AIR OPERATED CONTROL VALVE ASSY
1632	1MS-50-160	0	03/03/71	1" 600# AIR OPERATED CONTROL VALVE ASSY
1878	1MS-51-032-4	0	03/26/90	STM BREAK PROT & TURB PRESS SHT
1896	1MS-51-089		**/**/**	DELTA T/T AVG PROTECTION I
1897	1MS-51-101		**/**/**	DELTA T/T AVG PROTECTION II

DOC NUM	DOC NAME	REV	DATE	TITLE
1898	1MS-51-114		**/**/**	DELTA T/T AVG PROTECTION III
29	1MS-51-556	0	11/20/90	REACTOR VESSEL LEVEL SYSTEM
1879	1MS-51-642-1	0	07/17/81	PROC CONTROL SYSTEM SCHEMATIC DIAGRAM
1609	1MS-54-175-1	0	08/11/77	MOTOR DATA - RB PURGE SUPPLY
1658	1MS-54-175-14-1	0	05/23/77	MOTOR DATA - RB CHAR CLEAN UP
1619	1MS-54-175-2	0	05/23/77	MOTOR DATA - RB PURGE SUPPLY
1659	1MS-54-175-25-1	0	05/23/77	MOTOR DATA - DIESEL GEN RM A EXHAUST
1633	1MS-54-175-26-1	0	05/23/77	MOTOR DATA - CONTROL ACCESS EXHAUST
1174	1MS-54-175-27-1	0	05/23/77	MOTOR DATA - CONTROL ROOM EMERG FILTER SYSTEM
1623	1MS-54-175-6	0	04/12/77	MOTOR DATA - AUX BLDG CHAR
1629	1MS-54-175-7	0	04/12/77	MOTOR DATA - AUX BLDG CHAR
1631	1MS-54-175-9	0	04/12/77	MOTOR DATA - FUEL HANDELING BLDG EXHAUST
1664	1MS-55-155	0	06/04/82	FUEL PIPING DETAILS
728	1MS-94B-0003	3	**/**/**	COPES/VULCAN (QUICK CHANGE TRIM AIR OPERATED DIAPHRAGM CONTROL VALVES D-100)
647	1MS-94B-0004	0	**/**/**	CRANE CO (SEAL-LESS LEAKPROOF CANNED MOTOR PUMPS)
673	1MS-94B-0010	1	**/**/**	GENERAL ELECTRIC (STEAM TURBINE GEN - TURBINE SECTION)
1590	1MS-94B-0011	1	**/**/**	ITT GRINNELL (DIAPHRAGM VALVES)
660	1MS-94B-0025	1	**/**/**	PACIFIC PUMP CO (CHARGING/SAFETY INJECTION PUMP)
1882	1MS-94B-0052	0	**/**/**	AUXILARY BOILER SERVICE GUIDE
657	1MS-94B-0105	0	**/**/**	INGERSOL-RAND (CONDENSATE PUMPS)
1175	1MS-94B-0108	0	**/**/**	HATHAWAY INSTRUMENTS (FAULT RECORDING SYSTEM)
665	1MS-94B-0131	0	**/**/**	CRANE CO (REACTOR MAKEUP WATER PUMP GK,GKS,GP)
648	1MS-94B-0137	0	**/**/**	CRANE CO (CHEMPUMP-SERIES G COMPONENT COOLING WATER BOOSTER PUMP)
664	1MS-94B-0162	0	**/**/**	UNION PUMP CO (INSIDE TYPE DRA SEAL PUMP VLK)
678	1MS-94B-0279	4	**/**/**	INGERSOL-RAND (CENTRIFUGAL AIR COMPRESSORS)
682	1MS-94B-0301	0	**/**/**	UNION PUMP CO (HYDRO TEST PUMP TYPE-60)

DOC NUM	DOC NAME	REV	DATE	TITLE
650	1MS-94B-0302	1	**/**/**	INGERSOL-RAND (RESIDUAL HEAT REMOVAL PUMPS)
667	1MS-94B-0320	0	**/**/**	GOULD PUMPS (DIESEL GENERATOR SUMP PUMP MODELS 3171,3172,3173, & 3174)
656	1MS-94B-0343	0	**/**/**	HAZLETON PUMPS (HAZLETON TYPE VD & VS PUMPS)
674	1MS-94B-0368	0	**/**/**	BINGHAM-WILLAMETTE CO (CIRCULATING WATER PUMP TYPE SAFV)
1894	1MS-94B-0377	0	**/**/**	LOUIS ALLIS (LARGE MOTORS)
1893	1MS-94B-0386	0	**/**/**	GENERAL ELECTRIC (CUSTOM 8000 INDUCTION PUMPS)
675	1MS-94B-0402	6	**/**/**	RELIANCE ELECTRIC CO (REACTOR BUILDING FEEDPUMP COOLING UNITS)
1885	1MS-94B-0436	0	**/**/**	RELIANCE ELECTRIC CO (SWITCHES)
662	1MS-94B-0437	0	**/**/**	GOULD PUMPS (SERVICE WATER PUMPS)
1618	1MS-94B-0442	4	**/**/**	ANCHOR/DARLING (LARGE NUCLEAR POWER PLANT VALVES)
681	1MS-94B-0447	0	**/**/**	BINGHAM-WILLAMETTE CO (HS SERIES COMPONENT COOLING CLOSED CYCLE PUMP)
1054	1MS-94B-0485	0	**/**/**	GOULD-BROWN BOVERI (CONTROL POWER TRANSFORMER)
659	1MS-94B-0496	6	**/**/**	WESTINGHOUSE (REACTOR COOLANT PUMP MANUAL)
1697	1MS-94B-0520	0	**/**/**	YORK ELECTRO CONTROL CO (WASTE PROCESSING PANEL)
691	1MS-94B-0521	1	**/**/**	GOULD-BROWN BOVERI (INDOOR SECONDARY UNIT SUBSTATION)
1881	1MS-94B-0563	0	**/**/**	BENTLEY NEVADA PROBES AND PROXIMITORS
671	1MS-94B-0593	0	**/**/**	GOULD PUMPS (SERVICE WATER BOOSTER PUMP MODEL 3405L)
672	1MS-94B-0599	0	**/**/**	GOULD PUMPS (REACTOR BUILDING SPRAY PUMPS MODEL 34156)
655	1MS-94B-0711	1	**/**/**	JOY MFG CO (NUCLEAR AXIVANE FAN)
1596	1MS-94B-0761	0	**/**/**	FISHER CONTROLS (NUCLEAR CONTROL VALVES)
1604	1MS-94B-0772	2	**/**/**	ROCKWELL INTERNATION (ROCKWELL-EDWARD VALVES)
652	1MS-94B-0802	4	**/**/**	BAHNSON (AIR HANDLING UNITS)
1052	1MS-94B-0846	1	**/**/**	BAHNSON (DAMPER OPERATOR MAINTENANCE)
1598	1MS-94B-0865	4	**/**/**	ATWOOD & MORRILL (MAIN STEAM ISOLATION VALVE)
679	1MS-94B-0873	0	**/**/**	AIR INCORP (AIR COMPRESSOR MANUAL)

DOC NUM	DOC NAME	REV	DATE	TITLE
1942	1MS-94B-0898	0	**/**/**	CONTAINMENT HYDROGEN MONITOR TECHNICAL MANUAL
676	1MS-94B-0970	1	**/**/**	JOY MFG CO (AXIVANE FAN OPERATORS .ANDBOOK)
670	1MS-94B-1018	0	**/**/**	UNION PUMP CO (CHILLWATER PUMPS 50 HP)
669	1MS-94B-1019	1	**/**/**	UNION PUMP CO (CHILLWATER PUMPS 60 HP)
625	1MS-94B-1193	0	**/**/**	METEROLOGICAL INSTRUMENTS
1613	1MS-94B-1258	0	**/**/**	ANCHOR/DARLING (DOUBLE DISC TYPE GATE VALVES)
118	201-321	7	05/23/88	MCB FRONT VIEW XCP-6101 & 6102
119	201-322	7	05/23/88	MCB FRONT VIEW XCP-6103, 6104 & 6105
120	201-323	7	06/29/87	MCB FRONT VIEW XCP-6106 & 6107
121	201-324	6	12/02/88	MCB FRONT VIEW XCP-6108
122	201-325	9	09/16/87	MCB FRONT VIEW XCP-6109
123	201-326	5	09/14/88	MCB FRONT VIEW XCP-6110
124	201-327	9	06/29/87	MCB FRONT VIEW XCP-6111 & 6112
125	201-328	7	10/14/87	MCB FRONT VIEW XCP-6112
126	201-329	6	08/06/87	MCB FRONT VIEW XCP-6113
127	201-330	1	06/27/89	MCB FRONT VIEW XCP-6114
128	201-331	3	06/29/87	MCB FRONT VIEW XCP-6115
129	201-332	6	05/04/88	MCB FRONT VIEW XCP-6116
130	201-333	7	01/11/89	MCB FRONT VIEW XCP-6117
131	201-334	7	01/17/89	MCB FRONT VIEW XCP-6118
171	208-051		**/**/**	ELEMENTARY - HEATER DRAINS - HD
188	208-074		**/**/**	ELEMENTARY - NUCLEAR INSTRUMENTATION - NI
5	302-011	21	02/03/89	MAIN STEAM (NUCLEAR) - MS
6	302-012	16	10/11/87	MAIN STEAM (NON-NUCLEAR) - MS
7	302-014	7	05/12/82	MAIN & REHEAT STEAM (NON-NUCLEAR) - MS
8	302-031	9	07/11/86	MAIN STEAM DUMP SYSTEM - MS

DOC NUM	DOC NAME	REV	DATE	TITLE
9	302-041	9	05/12/86	EXTRACTION STEAM - EX
10	302-051	17	06/13/89	AUXILIARY STEAM
11	302-081	13	11/02/86	FEEDWATER (NON-NUCLEAR) - FW
12	302-082	12	07/11/87	FEEDWATER (NON-NUCLEAR) - FW
13	302-083	29	10/03/88	FEEDWATER (NUCLEAR) - FW
14	302-085	23	11/09/89	EMERGENCY FEEDWATER (NUCLEAR) - EF
15	302-101	35	10/23/89	CONDENSATE
16	302-102	14	12/16/86	CONDENSATE - AUXILIARY CONDENSERS & BLOWDOWN HEAT EXCHANGERS
18	302-111	19	08/06/89	HIGH PRESSURE HEATER DRIPS, VENTS, & RELIEFS
19	302-112	12	02/19/85	HIGH PRESSURE HEATER DRIPS, VENTS, & RELIEFS
20	302-113	9	07/24/87	LOW PRESSURE HEATER DRIPS, VENTS, & RELIEFS
21	302-121	7	09/18/84	MAIN STEAM DRAINS - MS
22	302-122	7	08/31/82	FEED PUMP START-UP, EXTRACTION & MISC STEAM DRAINS
23	302-123	10	04/09/89	MISC STEAM DRAINS
24	302-124	6	06/24/82	EXTRACTION STEAM DRAINS
26	302-131	14	07/11/86	CONDENSER AIR REMOVAL - AR
27	302-141	10	12/11/84	TURBINE GLAND STEAM
31	302-165	14	06/17/87	CONDENSATE POLISHING - CO
38	302-201	16	07/17/84	CIRCULATING WATER COOLING - CW
39	302-202	13	04/04/83	CIRCULATING WATER COOLING - CW
40	302-203	8	08/13/82	CIRCULATING WATER COOLING - CW
43	302-221	15	03/11/87	SERVICE WATER COOLING - SW
44	302-222	27	04/01/89	SERVICE WATER COOLING - SW
45	302-224	13	07/11/86	TURBINE ROOM CLOSED CYCLE COOLING WATER
46	302-231	23	05/17/89	FIRE SERVICE - FS

DOC NUM	DOC NAME	REV	DATE	TITLE
48	302-241	27	11/02/89	STATION SERVICE AIR
50	302-271	21	04/02/89	INSTRUMENT AIR - CAS
51	302-273	9	11/02/89	REACTOR BUILDING INSTRUMENT AIR SERVICES
53	302-281	10	08/04/82	FUEL OIL
55	302-301	5	12/20/82	GENERATOR GAS & VENTS
62	302-601	8	11/02/89	REACTOR COOLANT
63	302-602	13	01 11/90	REACTOR COOLANT SYSTEM - RCS
64	302-603	3	01/ 7/88	REACTOR COOLANT
65	302-604	4	01/07/88	REACTOR COOLANT
66	302-605	3	01/07/88	REACTOR COOLANT
68	302-611	22	10/11/89	COMPONENT COOLING - CC
69	302-612	18	09/18/89	COMPONENT COOLING SYSTEM INSIDE REACTOR BUILDING - CC
70	302-613	12	12/16/87	COMPONENT COOLING SYSTEM NON ESS EQUIPMENT COOLING - CC
72	302-641	5	03/15/87	RESIDUAL HEAT REMOVAL
73	302-651	26	05/31/89	SPENT FUEL COOLING - SF
74	302-661	22	12/17/88	REACTOR BUILDING SPRAY SYSTEM
75	302-671	4	07/21/86	CHEMICAL & VOLUME CONTROL - CVC
76	302-672	4	07/21/86	CHEMICAL & VOLUME CONTROL - CVC
77	302-673	6	03/10/89	CHEMICAL & VOLUME CONTROL - CVC
78	302-674	5	03/10/89	CHEMICAL & VOLUME CONTROL - CVC
79	302-675	9	01/23/90	CHEMICAL & VOLUME CONTROL - CVC
80	302-676	5	03/10/89	CHEMICAL & VOLUME CONTROL - CVC
81	302-677	3	07/31/86	CHEMICAL & VOLUME CONTROL - CVC
82	302-691	7	09/12/86	SAFETY INJECTION - SI
83	302-692	6	08/18/89	SAFETY INJECTION - SI
84	302-693	5	03/11/88	SAFETY INJECTION - SI

DOC NUM	DOC NAME	REV	DATE	TITLE
90	302-735	4	01/12/90	WASTE PROCESSING - WP
98	302-771	21B	08/24/89	NUCLEAR SAMPLING - SS
99	302-772	12	10/11/89	NORMAL & POST ACCIDENT SAMPLING - SS
100	302-781	19A	10/17/89	STEAM GENERATOR BLOWDOWN - SG
103	302-791	16	05/31/89	REACTOR MAKEUP WATER SYSTEM - RMW
106	302-821	19	11/29/85	REACTOR & AUXILIARY BUILDING SUMP PUMPS
109	302-824	3	09/21/82	REACTOR BUILDING COOLING UNIT DRAINS
111	302-841	18	10/02/87	CHILLED WATER - PUMP & CHILLER AREA - VU
112	302-842	12	10/14/88	CHILLED WATER - TO COOLING COILS A - VU
113	302-843	11	10/14/88	CHILLED WATER - TO COOLING COILS B - VU
115	302-851	21	01/09/87	INDUSTRIAL COOLING WATER - CI
116	302-852	6	12/20/83	CRDM COOLING WATER - AC
117	302-861	25	08/01/85	POST ACCIDENT HYDROGEN REMOVAL & ALTERNATE PURGE SYSTEM
224	804-660	11	06/12/87	CONTAINMENT ISOLATION PHASE A & SAFETY INJECTION ESF MONITOR LIGHTS
225	804-661	6	10/30/81	COLD LEG & HOT LEG RECIRCULATION ESF MONITOR LIGHTS
226	804-662	11	01/27/89	SAFETY INJECTION ESF MONITOR LIGHTS
227	804-663	4	10/30/81	CONTAINMENT ISOLATION PHASE B & REACTOR BUILDING SPRAY ESF MONITOR LIGHTS
228	804-664	6	11/07/86	WESTINGHOUSE SAFETY INJECTION GROUPS (1-3) ESF MONITOR LIGHTS
905	8756D01		**/**/**	BALANCE OF PLANT INSTRUMENTATION PROCESS CONTROL BLOCK DIAGRAM (1MS-51-221)
614	912-102	19	11/19/89	RB COOLING SYSTEM
280	912-103	18	03/24/89	PURGE EXHAUST SYSTEM
615	912-104	10	11/26/89	SECONDARY COMPARTMENT COOLING SYSTEM
622	912-105	8	11/17/89	REACTOR BUILDING REFUELING WATER SURFACE SYSTEM
616	912-106	10	11/20/89	CRDM SHROUD
276	912-115	15	08/10/89	AUX BLDG MAIN SUPPLY SYSTEM

DOC NUM	DOC NAME	R	DATE	TITLE
271	912-120	17	10/30/87	AUX BLDG HEPA EXHAUST SYSTEM
273	912-125	23	10/21/86	AUX BLDG CHARCOAL EXHAUST SYSTEM
278	912-130	17	11/13/87	AUX BLDG MAIN EXHAUST SYSTEM
275	912-131	18	11/16/82	FUEL HDLG CHAR EXHAUST AIR SUPPLY
296	912-134	10	10/08/87	DIESEL GENERATOR AREAS VENT SYSTEM
644	912-136	16	11/24/89	COMPUTER ROOM COOLING SYSTEM (CONTROL BLDG)
292	912-138	13	01/05/90	BAT RM CHGR RM & BOP CHRG VENT SYSTEM
643	912-140	25	11/23/89	CONTROL ROOM
646	912-144	15	11/20/89	CONTROL BLDG AIR HANDLING UNIT & CONDENSER
282	912-147	12	12/21/82	CONTROL BLDG CONTROLLED ACCESS EXHAUST
645	912-154	12	11/18/89	COMPUTER ROOM & SAS ROOM COOLING UNIT
694	912-155	15	10/08/87	SW INTR SCRN PP HSE BLDG VENT SYSTEM
291	912-158	11	12/21/82	IB GEN VENT & PUMP AREA CLG SYSTEM
1603	AB-436-K-07		**/**/**	LOCATION OFF EQUIPMENT TAG
1601	AB-436-N-07		**/**/**	LOCATION OFF EQUIPMENT TAG
1626	AB-463-J-07		**/**/**	LOCATION OFF EQUIPMENT TAG
1585	BOM		**/**/**	BILL OF MATERIAL
1053	BOM SM-03A		**/**/**	BILL OF MATERIAL SM-03A
1056	CB-482-G-09		**/**/**	LOCATION OFF EQUIPMENT TAG
1058	CB-482-G-12		**/**/**	LOCATION OFF EQUIPMENT TAG
1057	CB-482-G-13		**/**/**	LOCATION OFF EQUIPMENT TAG
1622	CB-482-H-11		**/**/**	LOCATION OFF EQUIPMENT TAG
1059	CB-482-H-13		**/**/**	LOCATION OFF EQUIPMENT TAG
692	CGSS-0264-SPE		08/15/90	INTER-OFFICE MEMO FROM SYSTEMS AND PERFORMANCE ENGINEERING
1592	CGSS-0294-SPE		11/11/90	DEGRADE/FAILURE AIR PRESSURE VALVES
229	CHAMPS		**/**/**	PLANT DATABASE INFORMATION (CHAMPS)

DOC NUM	DOC NAME	REV	DATE	TITLE
1880	DBD - CC SYSTEM	6	04/12/90	DESIGN BASIS DOC - COMPONENT COOLING
1887	DBD - RH SYSTEM	3	09/28/90	DESIGN BASIS DOC - RESIDUAL HEAT
1889	ENGINEERING ESTIMATE		**/**/**	ENGINEERING ESTIMATE
1663	FSAR		**/**/**	FINAL SAFETY ANALYSIS REPORT
2	GMP112.000		10/13/89	ELECTRICAL FEEDER LIST
1620	IB-427-F-06		**/**/**	LOCATION OFF EQUIPMENT TAG
1621	IB-427-F-07		**/**/**	LOCATION OFF EQUIPMENT TAG
1600	IB-436-G-04		**/**/**	LOCATION OFF EQUIPMENT TAG
1602	IB-436-L-03		**/**/**	LOCATION OFF EQUIPMENT TAG
700	ICP-100.018	4	07/03/89	SW BLDG MTR CONT CTR RM AMBIENT TEMP TE-9966
388	ICP-100.020	1	06/27/86	RB NARROW RANGE PRESSURE PT-8254
695	ICP-100.028	2	06/01/89	AIR HANDLING CONTROLLERS HK-9500
293	ICP-120.001	2	04/21/88	BTRS CHILLER FLOW FT-375
302	ICP-120.003	1	11/05/87	CHILLER SURGE TANK LEVEL LT-380
423	ICP-120.007	2	04/15/88	BTRS RETURN HDR TEMP TE-386
422	ICP-120.008	2	04/15/88	LETDOWN RHT HX OUT TEMP TE-381
1	ICP-130.001	5	05/01/90	VOLUME CONTROL TANK LEVEL LT-112
297	ICP-130.002	2	02/09/88	VOLUME CONTROL TANK LEVEL LT-115
231	ICP-130.004	4	04/11/90	BORIC ACID BLEND FLOW FT-113
232	ICP-130.005	4	06/28/90	CHARGING FLOW FT-122
236	ICP-130.006	4	02/02/89	LOW PRESSURE LETDOWN FLOW FT-150
243	ICP-130.007	3	10/04/88	TOTAL MAKEUP FLOW FT-168
349	ICP-130.008	3	09/27/90	VOLUME CONTROL TANK PRESSURE PT-117
350	ICP-130.011	3	08/30/89	LOW PRESSURE LETDOWN PRESSURE PT-145
412	ICP-130.012	3	02/10/88	VOLUME CONTROL TANK TEMP TE-116

DOC NUM	DOC NAME	REV	DATE	TITLE
413	ICP-130.014	2	02/10/88	EXCESS LETDOWN OUTLET TEMP TE-139
414	ICP-130.015	4	06/10/88	RHR HX OUTLET TEMP TE-140
415	ICP-130.016	2	05/26/88	LOW PRESSURE LETDOWN RELIEF LINE TEMP TE-141
416	ICP-130.017	4	01/25/90	DEMIN FLOW DIVERSION TE-143
448	ICP-130.020	2	06/20/89	BORIC ACID TANK TEMP TIS-107/TIS-109
452	ICP-140.003	4	04/20/90	DEAER STORAGE TANK WIDE RANGE LEVEL LT-3135
299	ICP-140.005	3	04/11/86	FW TURB A CONDENSER HOTWELL LEVEL LT-3142
300	ICP-140.006	2	03/13/86	FW TURB B CONDENSER HOTWELL LEVEL LT-3152
301	ICP-140.007	3	04/29/86	FW TURB C CONDENSER HOTWELL LEVEL LT-3162
455	ICP-140.008	2	07/31/89	CONDENSATE PUMP A DISCHARGE FLOW FT-3026
456	ICP-140.009	4	03/19/90	CONDENSATE PUMP B DISCHARGE FLOW FT-3036
463	ICP-140.010	2	07/31/89	CONDENSATE PUMP C DISCHARGE FLOW FT-3046
357	ICP-140.012	2	07/12/89	CONDENSATE A VACUUM PRESSURE PT-3006
358	ICP-140.013	2	06/12/86	CONDENSATE B VACUUM PRESSURE PT-3016
697	ICP-140.020	3	02/24/87	CONDENSATE TO DEAER FLOW CONTROLLER FC-3136
530	ICP-140.025	3	05/31/88	CONDENSATE FROM BLOWDOWN COOLER A TEMP TE-3063A
531	ICP-140.026	2	06/10/88	CONDENSATE FROM BLOWDOWN COOLER B TEMP TE-3063B
532	ICP-140.027	2	06/09/88	CONDENSATE FROM BLOWDOWN COOLER C TEMP TE-3063C
698	ICP-150.006	4	04/21/88	TURB OIL RESERVOIR TEMP TE-4211
430	ICP-150.007	2	02/02/89	GENERATOR H2 GAS TEMPERATURE TE-4265
431	ICP-150.008	2	04/28/87	ALTERNATOR COOLER TEMP TE-4270
286	ICP-160.007	4	05/24/88	CCW FROM RCP A THERMAL BARRIER FLOW FT-7138
287	ICP-160.008	4	01/18/89	CCW FROM RCP B THERMAL BARRIER FLOW FT-7158
288	ICP-160.009	4	01/18/89	CCW FROM RCP C THERMAL BARRIER FLOW FT-7178
283	ICP-160.014	3	05/29/89	CCW TO RHR HX A FLOW FT-7034
284	ICP-160.015	3	10/29/89	CCW TO RHR HX B FLOW FT-7044

DOC NUM	DOC NAME	REV	DATE	TITLE
285	ICP-160.017	2	11/08/85	CCW FROM EXCESS LETDOWN HEAT EXCHANGE FLOW FT-7106
386	ICP-160.028	3	04/13/90	CCW PUMP DISCH HDR A PRESS PT-7032
387	ICP-160.029	3	04/13/90	CCW PUMP DISCH HDR B PRESS PT-7042
323	ICP-160.030	5	12/02/88	CCW SURGE TANK LEVEL LT-7092
324	ICP-160.031	5	12/05/88	CCW SURGE TANK LEVEL LT-7094
444	ICP-160.032	2	05/11/88	CCW FROM EXCESS LETDOWN HEAT EXCHANGE TEMP TE-7108
249	ICP-195.010	4	03/31/89	EFW TO SG A FLOW FT-3531
250	ICP-195.011	5	06/29/90	EFW TO SG B FLOW FT-3541
730	ICP-195.021	2	01/22/90	EFW PUMP TURB SPEED ISI-13505/ISI-2036
745	ICP-205.013	3	07/11/89	EXTRACTION MS TO DEAER FOR PEGGING PRESSURE PT-2231
313	ICP-235.001	5	10/31/88	FW CONTROL VALVE FV-478
417	ICP-235.002	5	04/19/90	FW CONTROL VALVE FV-488
322	ICP-235.003	5	08/08/90	FW CONTROL VALVE FV-498
747	ICP-235.004	3	02/02/89	FW PUMP SPEED CONTROL PT-508
748	ICP-235.005	2	06/06/88	FW PUMP SPEED A CONTROL SC-509B
749	ICP-235.006	2	05/16/88	FW PUMP SPEED B CONTROL SC-509C
750	ICP-235.007	2	04/28/88	FW PUMP SPEED C CONTROL SC-509D
244	ICP-235.008	2	08/09/90	FW BOOSTER PUMP A DISCHARGE FLOW FT-3208
245	ICP-235.009	3	07/22/88	FW BOOSTER PUMP B DISCHARGE FLOW FT-3218
246	ICP-235.010	2	02/12/90	FW BOOSTER PUMP C DISCHARGE FLOW FT-3228
248	ICP-235.011	2	06/28/90	FW BOOSTER PUMP D DISCHARGE FLOW FT-3358
403	ICP-235.012	3	01/25/88	FW PUMP A DISCHARGE FLOW FT-3247
404	ICP-235.013	4	06/26/88	FW PUMP B DISCHARGE FLOW FT-3257
405	ICP-235.014	3	07/29/88	FW PUMP C DISCHARGE FLOW FT-3267
359	ICP-235.018	1	05/08/89	FW BOOSTER PUMP DISCHARGE HEADER PRESSURE PT-3239

DOC NUM	DOC NAME	REV	DATE	TITLE
418	ICP-235.025	5	06/22/89	FW TO SG A TEMP TE-3318
499	ICP-235.026	5	06/22/89	FW TO SG B TEMP TE-3328
420	ICP-235.027	4	10/30/89	FW TO SG C TEMP TE-3338
419	ICP-235.028	3	04/28/89	FW TO SG A TEMP TE-3322
500	ICP-235.029	3	04/28/89	FW TO SG B TEMP TE-3332
421	ICP-235.030	3	01/10/89	FW TO SG C TEMP TE-3342
497	ICP-235.031	3	08/18/89	FW HEADER TEMP TE-3307A
498	ICP-235.032	3	08/18/89	FW HEADER TEMP TE-3307B
329	ICP-235.033	3	08/26/88	FW BYPASS VALVE FV-3341
326	ICP-235.034	3	08/26/88	FW BYPASS VALVE FV-3331
756	ICP-235.035	2	07/19/88	FW BYPASS VALVE CONTROL QY-479
318	ICP-235.036	3	09/19/88	FW BYPASS VALVE FV-3321
763	ICP-240.019	4	07/05/89	BARTON INDICATING PRESS SWITCH/FLOW
543	ICP-240.087	1	04/28/88	GENERATOR H2 PURITY GAS ANALYZER
465	ICP-240.146	0	02/15/90	LEVEL-TROL CONTROLLERS (2502) (GENERIC)
408	ICP-245.003	1	03/31/89	STATOR COOLANT SYSTEM
447	ICP-265.001	3	12/11/89	INDUSTRIAL COOLER OUTLET TEMP TE-9213
701	ICP-265.004	2	02/01/90	RB TEMP TE-9205
487	ICP-290.006	5	04/13/90	SG A OUTLET PRESSURE PT-2000
355	ICP-290.007	6	06/08/90	SG B OUTLET PRESSURE PT-2010
488	ICP-290.008	5	01/18/89	SG C OUTLET PRESSURE PT-2020
480	ICP-300.001	4	01/10/90	RB SUMP A LEVEL LT-1963
298	ICP-300.002	4	03/05/90	RB SUMP B LEVEL LT-1964
424	ICP-335.001	3	01/29/88	REACTOR VESSEL FLANGE LEAK-OFF TE-431
433	ICP-340.007	3	08/23/88	PZR SURGE LINE TEMP TE-450
434	ICP-340.008	3	09/28/88	PZR SPRAY LINE TEMP TE-451

DOC NUM	DOC NAME	REV	DATE	TITLE
435	ICP-340.009	3	10/03/88	PZR SPRAY LINE TEMP
436	ICP-340.010	4	07/14/89	PZR LIQUID TEMP TE-453
437	ICP-340.011	3	08/29/88	PZR VAPOR TEMP TE-454
502	ICP-340.016	2	08/25/88	PZR RELIEF TANK TEMP TE-471
309	ICP-340.017	5	09/18/90	PZR RELIEF TANK LEVEL LT-470
490	ICP-340.018	3	08/26/88	PZR RELIEF TANK PRESSURE PT-472
351	ICP-340.019	4	01/24/89	RC PUMP C # 1 SEAL DELTA P PT-154
352	ICP-340.020	4	03/19/90	RC PUMP B # 1 SEAL DELTA P PT-155
353	ICP-340.021	4	06/09/89	RC PUMP A # 1 SEAL DELTA P PT-156
233	ICP-340.022	3	06/20/86	RC PUMP 3 SEAL WATER FLOW FT-124
234	ICP-340.023	3	05/14/86	RC PUMP 2 SEAL WATER FLOW FT-127
235	ICP-340.024	3	06/06/86	RC PUMP 1 SEAL WATER FLOW FT-130
237	ICP-340.025	3	03/09/89	RC PUMP C SEAL LEAK-OFF FLOW FT-154A
239	ICP-340.026	3	09/06/88	RC PUMP B SEAL LEAK-OFF FLOW FT-155A
241	ICP-340.027	3	03/10/89	RC PUMP A SEAL LEAK-OFF FLOW FT-156A
238	ICP-340.028	3	08/01/88	RC PUMP C SEAL LEAK-OFF FLOW FT-154B
240	ICP-340.029	3	07/22/88	RC PUMP B SEAL LEAK-OFF FLOW FT-155B
242	ICP-340.030	3	07/25/88	RC PUMP A SEAL LEAK-OFF FLOW FT-156B
344	ICP-340.037	2	08/02/90	LEVEL SWITCHES LS-406/407/408
327	ICP-341.001	3	03/20/90	REACTOR MAKEUP STORAGE TANK LEVEL LT-7603
328	ICP-341.002	3	06/14/90	REACTOR MAKEUP WATER STORAGE TANK LEVEL LT-7604
445	ICP-341.003	2	04/21/88	REACTOR MAKEUP STORAGE TANK TEMP TE-7601
446	ICP-341.004	2	04/19/88	REACTOR MAKEUP STORAGE TANK TEMP TE-7602
457	ICP-350.001	2	06/14/90	RHR PUMP A DISCHARGE PRESSURE PT-600A
491	ICP-350.002	2	11/14/89	RHR PUMP B DISCHARGE PRESSURE PT-600B

DOC NUM	DOC NAME	REV	DATE	TITLE
281	ICP-350.007	6	04/28/90	RHR HX A BYPASS FLOW FT-605A
474	ICP-350.008	6	04/19/90	RHR HX B BYPASS FLOW FT-605B
473	ICP-350.009	1	06/19/89	RHR PUMP DIFFERENTIAL PRESSURE FS-602A/B
305	ICP-365.003	3	10/17/88	PZR LEVEL L-409
426	ICP-365.007	3	05/04/90	DELTA T ROD INSERTION MONITORS T-409
411	ICP-365.009	5	06/28/90	T-TAVG CONTROL: ROD SPEED, POWER MISMATCH, STEAM DUMP TY-408
696	ICP-365.010	2	04/06/90	STEAM HEADER PRESSURE PT-464
539	ICP-395.020	3	12/08/87	ACCUMULATOR TANK A PRESSURE PT-921
492	ICP-395.021	4	02/28/90	ACCUMULATOR TANK B PRESSURE PT-925
494	ICP-395.022	3	08/30/89	ACCUMULATOR TANK C PRESSURE PT-929
390	ICP-395.023	4	01/19/90	ACCUMULATOR TANK A PRESSURE PT-923
493	ICP-395.024	4	02/08/90	ACCUMULATOR TANK B PRESSURE PT-927
495	ICP-395.025	4	02/16/90	ACCUMULATOR TANK C PRESSURE PT-931
481	ICP-395.026	3	10/04/88	ACCUMULATOR TANK A LEVEL LT-920
483	ICP-395.027	3	11/09/89	ACCUMULATOR TANK B LEVEL LT-924
485	ICP-395.028	3	01/10/90	ACCUMULATOR TANK C LEVEL LT-828
482	ICP-395.029	3	09/24/88	ACCUMULATOR TANK A LEVEL LT-922
484	ICP-395.030	3	09/24/88	ACCUMULATOR TANK B LEVEL LT-926
334	ICP-395.031	3	10/10/88	ACCUMULATOR TANK C LEVEL LT-930
476	ICP-395.047	5	03/13/90	RC PUMP UPPER AND LOWER BEARING FLOW FT-7263B
477	ICP-395.048	5	03/22/89	RC PUMP THERMAL BARRIER FLOW FT-7273A
303	ICP-400.001	2	01/12/90	SW POND WATER LEVEL LT-4418
304	ICP-400.002	2	02/09/90	SW POND WATER LEVEL LT-4458
265	ICP-400.007	0	11/16/89	SW FROM HVAC MECH WATER CHILLER A FLOW FT-4461
269	ICP-400.008	4	11/07/88	SW FROM HVAC MECH WATER CHILLER B FLOW FT-4491
266	ICP-400.009	3	11/19/88	SW FROM HVAC MECH WATER CHILLER C FLOW FT-4463

DOC NUM	DOC NAME	REV	DATE	TITLE
470	ICP-400.010	4	02/06/90	SW FROM HVAC MECH WATER CHILLER C FLOW FT-4493
268	ICP-400.013	1	09/21/88	SW POND RBCU RET HDR A INLET FLOW FT-4468
471	ICP-400.014	2	06/05/90	SW FROM RB HDR B AH HX FLOW FT-4498
362	ICP-400.015	2	11/27/89	SW PUMP A DISCH PRESSURE PT-4402
363	ICP-400.016	3	11/14/89	SW PUMP B DISCH PRESSURE PT-4422
364	ICP-400.018	3	03/15/90	SW PUMP C DISCH PRESSURE PT-4443
368	ICP-400.019	2	02/20/90	SW BOOSTER PUMP A DISCH PRESSURE PT-4523
370	ICP-400.020	2	09/25/90	SW BOOSTER PUMP B DISCH PRESSURE PT-4543
369	ICP-400.021	2	09/25/90	SW FROM RB HDR A AH HX PRESSURE PT-4528
371	ICP-400.022	3	10/13/89	SW FROM RM HDR B AH HX PRESSURE PT-4548
501	ICP-440.001	1	03/01/90	TURB RM CL CYCLE HX DISCH HDR TEMP TE-4606
699	ICP-440.002	0	04/16/79	EHC RESERVE HYD OIL TEMP TE-4637
409	ICP-440.003	1	08/10/83	TURB LUBE OIL
454	ICP-440.004	1	03/26/90	HYDRAULIC FLUID PUMP
347	ICP-440.005	0	04/09/85	TURB CLOSED CYCLE COOLING
627	ICP-440.010	0	11/15/85	TURB VIBRATION
479	ICP-451.003	3	06/05/90	RC DRAIN TANK LEVEL LT-1003
486	ICP-451.008	2	10/10/88	RC DRAIN TANK PRESSURE PT-1004
542	INSTRUMENT LIST		11/22/89	CHAMPS (INSTRUMENT LIST)
1586	MAINT ENG BOOK		**/**/**	MAINTENANCE ENGINEER BOOK
1161	NOT APPLICABLE		**/**/**	SOURCE DOCUMENT NOT REQUIRED
	NOT FOUND	NA	**/**/**	NOT FOUND
1051	PM #PO126116		**/**/**	REEL 3292 FRAME 2050
1588	PM #PO126173		**/**/**	REEL 3278 FRAME 1549
1903	PROCEDURES 10.2.02	2	05/01/84	NUCLEAR INSTRUMENTATION

DOC NUM	DOC NAME	REV	DATE	TITLE
1904	PROCEDURES 10.2.03	2	04/01/86	IN-CORE FLUX MAPPING
1905	PROCEDURES 10.2.04	0	07/01/83	CONTROL ROD DRIVE AND DRPI
1906	PROCEDURES 10.2.05	3	09/01/84	REACTOR COOLING SYSTEM
1907	PROCEDURES 10.2.06	2	04/01/85	REACTOR COOLANT PUMPS
1908	PROCEDURES 10.2.07	3	09/01/84	PRESSURIZER
1946	PROCEDURES 10.2.08	2	11/01/81	PRESSURIZER RELIEF TANK
1970	PROCEDURES 10.2.10	2	01/01/83	STEAM GENERATOR BLOWDOWN
1971	PROCEDURES 10.2.11	2	05/01/85	CHEMICAL AND VOLUME CONTROL
2044	PROCEDURES 10.2.14	2	03/01/85	COMPONENT COOLING SYSTEM
2045	PROCEDURES 10.2.15	2	12/01/82	CONTAINMENT SPRAY SYSTEM
2046	PROCEDURES 10.2.16	2	12/01/84	CONTAINMENT (VENT AND PURGE)
2047	PROCEDURES 10.2.17	1	12/01/82	REACTOR MAKEUP WATER
2048	PROCEDURES 10.2.18	1	08/01/83	RADIATION MONITORING SYSTEM
2049	PROCEDURES 10.2.19	1	03/01/83	NUCLEAR SAMPLING SYSTEM
2052	PROCEDURES 10.2.22	1	01/01/84	REACTOR AND AUX BUILDING SUMPS
2053	PROCEDURES 10.2.23	0	06/01/84	TEMPERATURE MONITORING SYSTEM
2055	PROCEDURES 10.2.25	1	05/01/85	ACOUSTIC LEAK MONITOR
2056	PROCEDURES 10.2.26	1	05/10/84	CORE COOLING MONITOR
2057	PROCEDURES 10.3.01	3	06/01/83	MAIN STEAM HEADER
2058	PROCEDURES 10.3.02	1	03/01/84	MAIN TURBINE
2059	PROCEDURES 10.3.03	2	12/01/84	TURBINE ELECTROHYDRAULIC CONTROL
2060	PROCEDURES 10.3.04	1	03/01/84	TURBINE SUPERVISORY INST
2061	PROCEDURES 10.3.05	2	12/01/84	CONDENSATE AND MAIN FEEDWATER
2062	PROCEDURES 10.3.06	1	01/01/84	FEEDWATER HEATER VENTS AND DRAINS
2063	PROCEDURES 10.3.07	2	07/01/84	EMERGENCY FEEDWATER
2064	PROCEDURES 10.3.08	1	03/01/84	AUXILIARY STEAM AND BOILER

DOC NUM	DOC NAME	REV	DATE	TITLE
2065	PROCEDURES 10.3.09	1	01/01/84	CIRCULATING WATER SYSTEM
2067	PROCEDURES 10.3.11	0	01/01/83	FIRE PROTECTION SYSTEM
2068	PROCEDURES 10.3.12	1	06/01/83	HEATING, VENTILATION AND AIR CONDITIONING
2069	PROCEDURES 10.3.13	3	11/01/85	MAIN GENERATOR
2070	PROCEDURES 10.3.14	3	11/01/85	DIESEL GENERATORS
2071	PROCEDURES 10.3.15	3	10/01/85	ELECTRICAL PLANT SYSTEM
2073	PROCEDURES 10.3.17	3	11/01/85	SYNCHRONIZER
2075	PROCEDURES 10.3.19	1	04/01/83	ENVIRONMENTAL MONITORING SYSTEM
2076	PROCEDURES 10.3.20	1	02/01/84	TURBINE GENERATOR AUXILARY
2077	PROCEDURES 10.3.21	2	03/01/84	SERVICE WATER SYSTEM
2079	PROCEDURES 10.3.23	1	01/01/83	LIQUID WASTE PROCESSING
2081	PROCEDURES 10.4.01	1	05/01/84	PLANT CONTROL SYSTEM
1950	PROCEDURES 10.4.02	1	09/01/83	PLANT PROTECTION LOGIC
1615	SDD		**/**/**	SYSTEM DESIGN DESCRIPTION
247	SDR 91-193		03/12/91	CIRCULATING WATER PUMPS AMPS WRONG
252	SDR 91-194		03/12/91	CONDENSATE PUMPS AMPS WRONG
230	SDR 91-201		03/12/91	SERVICE WATER PUMP AMPS WRONG
540	SETPOINT LIST		11/22/89	CHAMPS (SETPOINT LIST)
1888	START UP MANUAL		**/**/**	START UP MANUAL - COMPONENT COOLING WATER
628	STP-105.002	5	04/01/89	CVC VALVE OPERABILITY TEST
601	STP-105.003	10	04/01/89	SAFETY INJECTION VALVE OPERABILITY TEST
603	STP-105.005	7	04/01/89	RHR VALVE OPERABILITY TEST
602	STP-112.003	5	03/28/89	RB SPRAY VALVE OPERABILITY TEST
633	STP-120.004	11	01/24/90	EFW VALVE OPERABILITY TEST
599	STP-121.002	9	04/01/89	MAIN STEAM OPERABILITY TEST VALVE

DOC NUM	DOC NAME	REV	DATE	TITLE
600	STP-122.002	11	06/26/90	CCW PUMP TEST
632	STP-123.002	11	01/23/90	SW PUMP TEST
630	STP-123.003A	0	04/02/89	SW TRAIN A VALVE OPERABILITY TEST
631	STP-123.003B	0	04/01/89	SW TRAIN B VALVE OPERABILITY TEST
637	STP-127.001	5	03/17/89	PZR BLOCK VALVE OPERABILITY TEST
635	STP-129.002	6	07/15/88	HVAC CHILLED WATER SYSTEM VALVE OPERATION TEST
629	STP-130.003	3	11/20/89	VALVE OPERABILITY TESTING (MODES 1, 2, AND 3)
595	STP-130.004	5	04/01/89	VALVE OPERABILITY TESTING (MODE 4)
612	STP-130.005A	0	04/01/89	AC VALVE OPERABILITY TESTING (MODE 5)
605	STP-130.005B	0	04/01/89	AIR HANDLING VALVE OPERABILITY TESTING (MODE 5)
606	STP-130.005C	0	04/01/89	CCW VALVE OPERABILITY TESTING (MODE 5)
607	STP-130.005D	0	04/02/89	CCW VALVE OPERABILITY TESTING (MODE 5)
608	STP-130.005E	0	04/01/89	EFW VALVE OPERABILITY TEST (MODE 5)
609	STP-130.005G	0	04/02/89	INSTRUMENT AIR VALVE OPERABILITY TESTING (MODE 5)
610	STP-130.005H	0	04/02/89	SAFETY INJECTION VALVE OPERABILITY TESTING (MODE 5)
611	STP-130.005I	0	04/02/89	RB SPRAY VALVE OPERABILITY TESTING (MODE 5)
596	STP-136.001	4	03/29/89	SG BLOWDOWN VALVE OPERABILITY TEST
597	STP-138.001	6	04/01/89	POST ACCIDENT HYDROGEN REMOVAL VALVE OPERABILITY TEST
634	STP-140.001	5	03/29/89	RB AND AUX BLDG NUCLEAR DRAINS VALVE OPERABILITY TEST
638	STP-142.001	6	01/18/90	REACTOR COOLANT SYSTEM VALVE OPERATION TEST
639	STP-144.001	5	03/09/89	NUCLEAR SAMPLING VALVE OPERATION TEST
636	STP-145.001	5	04/01/89	WASTE PROCESSING VALVE OPERATION TEST
604	STP-146.003	1	03/28/89	REACTOR MAKEUP WATER VALVE OPERABILITY TEST
598	STP-148.001	6	04/01/89	FW VALVE OPERATION TEST
642	STP-170.003	1	01/03/90	FIRE SERVICE VALVE OPERABILITY TEST
513	STP-300.030	4	09/22/88	RB SUMP LEAK DETECTOR LEVEL CALIBRATION LE-1962

Source Documents used in CMS as of 03/21/91

DOC NUM	DOC NAME	REV	DATE	TITLE
588	STP-302.001	6	04/04/90	DELTA T - TAVG PROTECTION LOOP 1 OPERATION TEST NO1 - CS TEST I
561	STP-302.002	6	04/04/90	DELTA T - TAVG PROTECTION LOOP 2 OPERATION TEST NO1 - CS TEST I
562	STP-302.003	7	04/05/90	DELTA T - TAVG PROTECTION LOOP 3 OPERATION TEST NO1 - CS TEST I
557	STP-302.007	3	03/03/87	PZR LEVEL OPERATION TEST LT-459
518	STP-310.001	5	11/01/88	NIS SOURCE RANGE CALIBRATION N31
519	STP-310.002	6	10/29/88	NIS SOURCE RANGE CALIBRATION N32
520	STP-310.003	5	12/07/88	NIS INTERMEDIATE RANGE CALIBRATION N35
521	STP-310.004	6	11/09/88	NIS INTERMEDIATE RANGE CALIBRATION N36
522	STP-310.005	7	05/06/87	NIS POWER RANGE CALIBRATION N41
523	STP-310.006	5	05/06/87	NIS POWER RANGE CALIBRATION N42
524	STP-310.007	5	05/06/87	NIS POWER RANGE CALIBRATION N43
525	STP-310.008	4	05/06/87	NIS POWER RANGE CALIBRATION N44
1285	STP-310.016	0	01/29/90	FLUX DEVIATION AND MISC CONTROL AND INDICATION DRAWER CALIBRATION N50
558	STP-340.001	6	11/11/87	RC LOOP A WIDE RANGE HOT LEG TEMP CALIBRATION TE-413
427	STP-340.004	6	11/11/87	RC LOOP A WIDE RANGE COLD LEG TEMP CALIBRATION TE-410
365	STP-340.005	4	05/01/87	PORV ACTUATION CHANNEL CALIBRATION PT-444,PT-445
439	STP-340.010	2	06/27/89	PZR RELIEF LINE TEMP CALIBRATION TE-463
440	STP-340.012	1	03/23/89	PZR SAFETY VALVE LINE TEMP CALIBRATION TE-465
441	STP-340.013	2	03/28/89	PZR SAFETY VALVE LINE TEMP CALIBRATION TE-467
442	STP-340.014	2	01/16/89	PZR SAFETY VALVE LINE TEMP CALIBRATION TE-469
512	STP-342.003	2	07/15/88	RBCU CONDENSATE FLOW FS-1900A/B
590	STP-345.002	6	04/04/90	DELTA T - TAVG PROTECTION LOOP 2 CALIBRATION NO 1 - CS TEST II
360	STP-345.004	5	06/22/89	RC LOOP WIDE RANGE PRESSURE CALIBRATION PT-402
361	STP-345.005	5	07/13/89	RC LOOP WIDE RANGE PRESSURE CALIBRATION PT-403
255	STP-345.006	4	05/26/88	RC LOOP A FLOW CALIBRATION FT-414

DOC NUM	DOC NAME	REV	DATE	TITLE
258	STP-345.007	5	05/26/88	RC LOOP B FLOW CALIBRATION FT-424
262	STP-345.008	5	04/18/88	RC LOOP C FLOW CALIBRATION FT-434
256	STP-345.009	5	04/15/88	RC LOOP A FLOW CALIBRATION FT-415
259	STP-345.010	5	05/31/88	RC LOOP B FLOW CALIBRATION FT-425
263	STP-345.011	5	06/10/88	RC LOOP C FLOW CALIBRATION FT-435
257	STP-345.012	5	05/24/88	RC LOOP A FLOW CALIBRATION FT-416
260	STP-345.013	6	05/26/88	RC LOOP B FLOW CALIBRATION FT-426
264	STP-345.014	5	05/24/88	RC LOOP C FLOW CALIBRATION FT-436
458	STP-345.015	3	11/07/88	PZR PRESSURE CALIBRATION PT-455
459	STP-345.016	3	11/07/88	PZR PRESSURE CALIBRATION PT-456
460	STP-345.017	2	11/07/88	PZR PRESSURE CALIBRATION PT-457
307	STP-345.019	2	10/01/86	PZR LEVEL CALIBRATION LT-460
308	STP-345.020	2	10/20/86	PZR LEVEL CALIBRATION LT-461
310	STP-345.021	3	11/03/88	SG A NARROW RANGE LEVEL CALIBRATION LT-474
314	STP-345.022	3	11/03/88	SG B NARROW RANGE LEVEL CALIBRATION LT-484
319	STP-345.023	3	11/04/88	SG C NARROW RANGE LEVEL CALIBRATION LT-494
311	STP-345.024	3	11/03/88	SG A NARROW RANGE LEVEL CALIBRATION LT-475
315	STP-345.025	4	11/03/88	SG B NARROW RANGE LEVEL CALIBRATION LT-485
320	STP-345.026	4	11/03/88	SG C NARROW RANGE LEVEL CALIBRATION LT-495
312	STP-345.027	5	11/04/88	SG A NARROW RANGE LEVEL CALIBRATION LT-476
317	STP-345.028	4	11/03/88	SG B NARROW RANGE LEVEL CALIBRATION LT-486
321	STP-345.029	3	11/03/88	SG C NARROW RANGE LEVEL CALIBRATION LT-496
294	STP-345.030	3	09/27/88	BORIC ACID TANK A LEVEL CALIBRATION LT-106
295	STP-345.031	3	02/09/88	BORIC ACID TANK B LEVEL CALIBRATION LT-161
366	STP-345.034	4	12/01/88	TURB FIRST STAGE PRESSURE CALIBRATION PT-446
367	STP-345.035	4	12/01/88	TURB FIRST STAGE PRESSURE CALIBRATION PT-447

DOC NUM	DOC NAME	REV	DATE	TITLE
325	STP-370.001	3	09/22/88	SODIUM HYDROXIDE STORAGE TANK LEVEL CALIBRATION LT-7356
461	STP-370.002	3	09/24/90	SODIUM HYDROXIDE STORAGE TANK LEVEL CALIBRATION LT-7358
289	STP-370.003	3	09/25/90	RB SPRAY PUMP A DISCHARGE FLOW CALIBRATION FT-7368
290	STP-370.004	3	09/30/88	RB SPRAY PUMP B DISCHARGE FLOW CALIBRATION FT-7378
335	STP-375.001	4	09/14/88	RWST LEVEL CALIBRATION LT-990
337	STP-375.002	5	11/01/88	RWST LEVEL CALIBRATION LT-991
338	STP-375.003	4	10/17/88	RWST LEVEL CALIBRATION LT-992
339	STP-375.004	4	10/07/88	RWST LEVEL CALIBRATION LT-993
377	STP-395.001	3	12/03/87	SG A STEAM PRESSURE CALIBRATION PT-475
380	STP-395.002	4	12/01/87	SG B STEAM PRESSURE CALIBRATION PT-485
383	STP-395.003	4	11/10/87	SG C STEAM PRESSURE CALIBRATION PT-495
272	STP-395.004	4	05/23/88	SG A STEAM/FW FLOW CALIBRATION FT-474,FT-477
538	STP-395.005	5	04/12/88	SG B STEAM/FW FLOW CALIBRATION FT-484,FT-487
511	STP-395.006	5	04/13/88	SG C STEAM/FW FLOW CALIBRATION FT-494,FT-497
509	STP-395.007	4	05/20/88	SG A STEAM/FW FLOW CALIBRATION FT-475,FT-476
277	STP-395.008	4	05/26/88	SG B STEAM/FW FLOW CALIBRATION FT-485,FT-486
279	STP-395.009	4	05/18/88	SG C STEAM/FW FLOW CALIBRATION FT-495,FT-496
396	STP-395.016	3	04/24/87	CONTAINMENT PRESSURE CALIBRATION PT-950,PT-954A
535	STP-395.017	3	04/22/87	CONTAINMENT PRESSURE CALIBRATION PT-951
398	STP-395.018	4	05/20/87	CONTAINMENT PRESSURE CALIBRATION PT-952
399	STP-395.019	3	05/18/87	CONTAINMENT PRESSURE CALIBRATION PT-953
462	STP-395.036	4	05/31/88	SG A STEAM PRESSURE CALIBRATION PT-476
534	STP-395.037	4	02/10/88	SG B STEAM PRESSURE CALIBRATION PT-486
384	STP-395.038	4	06/06/88	SG C STEAM PRESSURE CALIBRATION PT-496
376	STP-395.039	5	05/26/88	SG A STEAM PRESSURE CALIBRATION PT-474

DOC NUM	DOC NAME	REV	DATE	TITLE
379	STP-395.040	4	06/15/88	SG B STEAM PRESSURE CALIBRATION PT-484
382	STP-395.041	4	05/31/88	SG C STEAM PRESSURE CALIBRATION PT-494
507	STP-396.001	3	11/08/88	EFW TO SG A FLOW CALIBRATION FT-3561
253	STP-396.002	3	11/01/88	EFW TO SG B FLOW CALIBRATION FT-3571
254	STP-396.003	3	10/17/88	EFW TO SG C FLOW CALIBRATION FT-3581
541	STP-396.007	5	09/12/90	EFW PUMP SUCTION PRESSURE CALIBRATION PT-3632
526	STP-396.008	3	07/14/88	EFW PUMP SUCTION PRESSURE CALIBRATION PT-3633
527	STP-396.009	3	06/13/88	EFW PUMP SUCTION PRESSURE CALIBRATION PT-3634
528	STP-396.010	3	06/07/90	EFW PUMP SUCTION PRESSURE CALIBRATION PT-3635
267	STP-399.001	4	12/14/88	SW BOOSTER PUMP A DISCHARGE FLOW CALIBRATION FT-4466
270	STP-399.002	4	10/16/87	SW BOOSTER PUMP B DISCHARGE FLOW CALIBRATION FT-4496
594	STP-401.002	6	09/15/88	MAIN STEAM LINE CODE SAFETY VALVES ASME SECTION XI TEST
1614	TB-424-F-05		**/**/**	LOCATION OFF EQUIPMENT TAG
1597	TB-436-F-01		**/**/**	LOCATION OFF EQUIPMENT TAG
1593	TB-463-F-04		**/**/**	LOCATION OFF EQUIPMENT TAG
1744	VCS-IFC00478-FW	0	01/13/89	INST LOOP DIAG - SGA FW CONTROL
1748	VCS-IFC00488-FW	0	01/13/89	INST LOOP DIAG - SGA FW CONTROL
1752	VCS-IFC00498-FW	0	01/05/89	INST LOOP DIAG - SGB FW CONTROL
1901	VCS-IFT00110-CS	0	04/07/86	INST LOOP DIAG - BA EMERG BYPASS SYSTEM
1667	VCS-IFT00113-CS	0	06/04/87	INST LOOP DIAG - BA TO BLENDER FLOW
1668	VCS-IFT00122-CS	0	08/29/86	INST LOOP DIAG - CHARGING FLOW
1669	VCS-IFT00124-CS	0	05/29/86	INST LOOP DIAG - RCP C SEAL WATER INJECTION FLOW
1670	VCS-IFT00127-CS	0	11/26/85	INST LOOP DIAG - RCP B SEAL WATER INJECTION FLOW
1671	VCS-IFT00130-CS	0	01/06/86	INST LOOP DIAG - RCP A SEAL WATER INJECTION FLOW
1672	VCS-IFT00150-CS	0	07/29/86	INST LOOP DIAG - LOW PRESSURE LETDOWN FLOW
1673	VCS-IFT00154-CS	1	08/30/89	INST LOOP DIAG - RCP C SEAL LEAKOFF FLOW

DOC NUM	DOC NAME	REV	DATE	TITLE
1674	VCS-IFT00155-CS	1	08/30/89	INST LOOP DIAG - RCP B SEAL LEAKOFF FLOW
1675	VCS-IFT00156-CS	1	08/30/89	INSY LOOP DIAG - RCP A SEAL LEAKOFF FLOW
1676	VCS-IFT00168-CS	0	07/31/87	INST LOOP DIAG - TOTAL MU FLOW
1909	VCS-IFT00375-TR	0	08/22/88	INST LOOP DIAG - BTRS CHILLER FLOW
1910	VCS-IFT00385-TR	0	07/18/88	INST LOOP DIAG - BTRS RETURN HEADER FLOW
1677	VCS-IFT00414-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP A GROUP I
1678	VCS-IFT00415-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP A GROUP II
1679	VCS-IFT00416-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP A GROUP III
1680	VCS-IFT00424-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP B GROUP I
1681	VCS-IFT00425-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP B GROUP II
1682	VCS-IFT00426-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP B GROUP III
1683	VCS-IFT00434-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP C GROUP I
1684	VCS-IFT00435-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP C GROUP II
1685	VCS-IFT00436-RC	1	10/09/89	INST LOOP DIAG - RC FLOW LOOP C GROUP III
1686	VCS-IFT00474-MS	0	02/27/90	INST LOOP DIAG - SGA STEAM FLOW
1687	VCS-IFT00475-MS	0	02/27/90	INST LOOP DIAG - SGA STEAM FLOW
1688	VCS-IFT00476-FW	1	04/07/89	INST LOOP DIAG - SGA FW FLOW
1689	VCS-IFT00477-FW	2	01/30/89	INST LOOP DIAG - SGA FW FLOW
1690	VCS-IFT00484-MS	0	03/14/90	INST LOOP DIAG - SGB STEAM FLOW
1691	VCS-IFT00485-MS	0	07/26/89	INST LOOP DIAG - SGB STEAM FLOW
1692	VCS-IFT00486-FW	2	01/30/89	INST LOOP DIAG - SGB FW FLOW
1693	VCS-IFT00487-FW	0	12/07/88	INST LOOP DIAG - SGB FW FLOW
1694	VCS-IFT00494-MS	0	03/14/90	INST LOOP DIAG - SGC STEAM FLOW
1696	VCS-IFT00495-MS	0	08/21/89	INST LOOP DIAG - SGC STEAM FLOW
1695	VCS-IFT00496-FW	1	01/30/89	INST LOOP DIAG - SGC FW FLOW

DOC NUM	DOC NAME	REV	DATE	TITLE
1698	VCS-IFT00497-FW	1	01/30/89	INST LOOP DIAG - SGC FW FLOW
1699	VCS-IFT00605A-RH	1	09/11/90	INST LOOP DIAG - RHX BYPASS FLOW
1700	VCS-IFT00605B-RH	1	09/11/90	INST LOOP DIAG - RHX BYPASS FLOW
1911	VCS-IFT00940-SI	0	10/09/86	INST LOOP DIAG - SI HEADER FLOW
1912	VCS-IFT00943-SI	0	10/16/86	INST LOOP DIAG - SI HEADER FLOW
1913	VCS-IFT03009-CO	0	01/13/89	INST LOOP DIAG - CONDENSATE TANK TO CONDENSER FLOW
1914	VCS-IFT03019-CO	0	03/17/89	INST LOOP DIAG - MU TO CONDENSATE TANK FLOW
1915	VCS-IFT03026-CO	0	10/20/88	INST LOOP DIAG - CONDENSATE PMP A DISCHARGE FLOW
1916	VCS-IFT03036-CO	0	01/03/89	INST LOOP DIAG - CONDENSATE PMP B DISCHARGE FLOW
1917	VCS-IFT03046-CO	0	02/15/89	INST LOOP DIAG - CONDENSATE PMP C DISCHARGE FLOW
1918	VCS-IFT03061-CO	0	03/23/90	INST LOOP DIAG - SG BD HX CO FLOW
1919	VCS-IFT03120-CO	0	03/14/90	INST LOOP DIAG - CONDENSATE TO DEAER FLOW
1701	VCS-IFT03203-FW	0	02/14/90	INST LOOP DIAG - FW BOOSTER PMP A DISCHARGE FLOW
1702	VCS-IFT03213-FW	0	02/14/90	INST LOOP DIAG - FW BOOSTER PMP B DISCHARGE FLOW
1703	VCS-IFT03223-FW	0	02/14/90	INST LOOP DIAG - FW BOOSTER PMP C DISCHARGE FLOW
1920	VCS-IFT03247-FW	0	02/22/90	INST LOOP DIAG - MAIN FWP A DISCHARGE FLOW
1921	VCS-IFT03257-FW	0	02/14/90	INST LOOP DIAG - MAIN FWP B DISCHARGE FLOW
1922	VCS-IFT03267-FW	0	02/14/90	INST LOOP DIAG - MAIN FWP C DISCHARGE FLOW
1923	VCS-IFT03310-FW	0	02/27/90	INST LOOP DIAG - TOTAL FW FLOW TO SG
1924	VCS-IFT03323-FW	0	02/27/90	INST LOOP DIAG - SGA FW FORWARD FLUSH LINE FLOW
1926	VCS-IFT03333-FW	0	03/22/89	INST LOOP DIAG - SGB FW FORWARD FLUSH LINE FLOW
1925	VCS-IFT03343-FW	0	03/22/89	INST LOOP DIAG - SGC FW FORWARD FLUSH LINE FLOW
1704	VCS-IFT03358-FW	0	01/10/90	INST LOOP DIAG - FW BOOSTER PMP D DISCHARGE FLOW
1927	VCS-IFT03525-EF	0	02/06/89	INST LOOP DIAG - TURBINE DRIVEN EMERG FWP DISCHARGE FLOW
1705	VCS-IFT03531-EF	0	03/22/90	INST LOOP DIAG - EFW TO SGA FLOW (TRAIN B)
1706	VCS-IFT03541-EF	0	02/14/90	INST LOOP DIAG - EFW TO SGB FLOW (TRAIN B)

DOC NUM	DOC NAME	REV	DATE	TITLE
1707	VCS-IFT03551-EF	0	04/20/89	INST LOOP DIAG - EFW TO SGC FLOW (TRAIN B)
1928	VCS-IFT03561A-EF	0	06/27/88	INST LOOP DIAG - EFW TO SGA FLOW (TRAIN B)
1708	VCS-IFT03561-EF	0	06/27/88	INST LOOP DIAG - EFW TO SGA FLOW (TRAIN B)
1929	VCS-IFT03571A-EF	0	03/22/90	INST LOOP DIAG - EFW TO SGB FLOW (TRAIN B)
1709	VCS-IFT03571-EF	0	03/22/90	INST LOOP DIAG - EFW TO SGA FLOW
1930	VCS-IFT03581A-EF	0	03/22/89	INST LOOP DIAG - EFW TO SGC FLOW (TRAIN B)
1710	VCS-IFT03581-EF	0	03/22/89	INST LOOP DIAG - EFW TO SGC FLOW (TRAIN B)
1711	VCS-IFT04461-SW	0	05/02/89	INST LOOP DIAG - SW FLOW FROM HVAC CHILLER A
1712	VCS-IFT04463-SW	0	02/28/90	INST LOOP DIAG - SW FLOW FROM HVAC CHILLER C (TRAIN A)
1713	VCS-IFT04466-SW	0	02/14/90	INST LOOP DIAG - SW BOOSTER PMP A DISCHARGE FLOW
1714	VCS-IFT04468-SW	0	11/23/88	INST LOOP DIAG - SW FROM RBCU A HEADER FLOW
1715	VCS-IFT04491-SW	0	05/16/89	INST LOOP DIAG - SW FLOW FROM HVAC CHILLER B
1716	VCS-IFT04493-SW	0	06/01/85	INST LOOP DIAG - SW FLOW FROM HVAC CHILLER C (TRAIN B)
1717	VCS-IFT04496-SW	0	05/16/89	INST LOOP DIAG - SW BOOSTER PMP B DISCHARGE FLOW
1718	VCS-IFT04498-SW	0	01/03/89	INST LOOP DIAG - SW FROM RBCU B HEADER FLOW
1719	VCS-IFT07034-CC	0	03/14/90	INST LOOP DIAG - CCW TO RHR HX A FLOW
1720	VCS-IFT07044-CC	0	08/15/89	INST LOOP DIAG - CCW TO RHR HX B FLOW
1721	VCS-IFT07106-CC	0	02/27/89	INST LOOP DIAG - CCW FROM EXCESS LETDOWN HX FLOW
1722	VCS-IFT07138-CC	0	02/22/89	INST LOOP DIAG - CCW FROM RCP A THERMAL BARRIER FLOW
1723	VCS-IFT07158-CC	0	02/14/89	INST LOOP DIAG - CCW FROM RCP B THERMAL BARRIER FLOW
1724	VCS-IFT07178-CC	0	01/03/89	INST LOOP DIAG - CCW FROM RCP C THERMAL BARRIER FLOW
1725	VCS-IFT07263A-CC	0	11/23/88	INST LOOP DIAG - RCP A/B/C UPPER & LOWER BEARING FLOW
1931	VCS-IFT07263B-CC	0	09/07/88	INST LOOP DIAG - RCP A/B/C UPPER & LOWER BEARING FLOW
1726	VCS-IFT07273A-CC	0	03/23/88	INST LOOP DIAG - RCP A/B/C THERMAL BARRIER FLOW
1932	VCS-IFT07273B-CC	0	01/05/89	INST LOOP DIAG - RCP A/B/C THERMAL BARRIER FLOW

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1727	VCS-IFT07368-SP	0	01/30/89	INST LOOP DIAG - RB SPRAY PMP A DISCHARGE FLOW
1728	VCS-IFT07378-SP	0	01/18/89	INST LOOP DIAG - RB SPRAY PMP B DISCHARGE FLOW
1933	VCS-IFT07449-SF	0	02/27/89	INST LOOP DIAG - SF HX A OUTLET FLOW
1934	VCS-IFT07459-SF	0	03/17/89	INST LOOP DIAG · SF HX B OUTLET FLOW
1935	VCS-IFT08251-HR	0	02/15/89	INST LOOP DIAG - POST ACCIDENT H2 REMOVAL TO PURGE EXHAUST STACK FLOW
1936	VCS-IFT08252-HR	0	02/27/90	INST LOO? DIAG - POST ACCIDENT H2 REMOVAL TO PURGE EXHAUST STACK FLOW
1937	VCS-IFT09405-AH	0	04/03/89	INST LOOP DIAG - CONTROL ROOM OUTSIDE AIR INTAKE FLOW A
1938	VCS-IFT09455-AH	0	03/14/90	INST LOOP DIAG - CONTROL ROOM OUTSIDE AIR INTAKE FLOW B
1939	VCS-IFT09492A-AH	0	04/03/89	INST LOOP DIAG - RELAY ROOM OUTSIDE AIR INTAKE FLOW A
1940	VCS-IFT09492B-AH	0	04/03/89	INST LOOP DIAG - RELAY ROOM OUTSIDE AIR INTAKE FLOW B
1941	VCS-IFT09502A-AH	0	04/03/89	INST LOOP DIAG - COMPUTER ROOM OUTSIDE AIR INTAKE A
1943	VCS-IFT09560-AH	0	03/14/90	INST LOOP DIAG - SAS/CPU-1.2 OUTSIDE AIR INTAKE FLOW
1944	VCS-IFT09951A-AH	0	04/20/89	INST LOOP DIAG - SW BLDG OUTSIDE AIR INTAKE A FLOW
1945	VCS-IFT09951B-AH	0	04/03/89	INST LOOP DIAG - SW BLDG OUTSIDE AIR INTAKE B FLOW
1732	VCS-ILT00106-CS	0	06/18/87	INST LOOP DIAG - BA TANK A LEVEL
1733	VCS-ILT00108-CS	0	06/09/87	INST LOOP DIAG - BA TANK B LEVEL
1734	VCS-ILT00112-CS	1	05/25/88	INST LOOP DIAG - VOLUME CONTROL TANK LEVEL
1735	VCS-ILT00115-CS	1	05/25/88	INST LOOP DIAG - VOLUME CONTPOL TANK LEVEL
1947	VCS-ILT00161-CS	0	06/10/87	INST LOOP DIAG - BA TANK A LEVEL
1948	VCS-ILT00163-CS	0	06/04/87	INST LOOP DIAG - BA TANK B LEVEL
1736	VCS-ILT00380-TR	0	01/05/89	INST LOOP DIAG - BORON CHILLER SURGE TANK LEVEL
1737	VCS-ILT00459-RC	0	10/06/89	INST LOOP DIAG - PRZR LEVEL XMTR
1738	VCS-ILT00460-RC	3	10/09/89	INST LOOP DIAG - PRZR LEVEL XMTR
1739	VCS-ILT00461-RC	3	10/09/89	INST LOOP DIAG - PRZR LEVEL XMTR
1949	VCS-ILT00462-RC	0	03/03/88	INST LOOP DIAG - PRZR COLD LEVEL XMTR
1740	VCS-ILT00470-RC	1	02/24/88	INST LOOP DIAG - PRZR RELIEF TANK LEVEL

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1741	VCS-ILT00474-MS	1	10/09/89	INST LOOP DIAG - SGA NARROW RANGE LEVEL PROT I
1742	VCS-ILT00475-MS	2	10/09/89	INST LOOP DIAG - SGA NARROW RANGE LEVEL PROT II
1743	VCS-ILT00476-MS	1	10/09/89	INST LOOP DIAG - SGA NARROW RANGE LEVEL PROT III
1745	VCS-ILT00484-MS	1	10/09/89	INST LOOP DIAG - SGB NARROW RANGE LEVEL PROT I
1746	VCS-ILT00485-MS	1	10/09/89	INST LOOP DIAG - SGB NARROW RANGE LEVEL PROT II
1747	VCS-ILT00486-MS	1	10/09/89	INST LOOP DIAG - SGB NARROW RANGE LEVEL PROT III
1749	VCS-ILT00494-MS	2	10/09/89	INST LOOP DIAG - SGC NARROW RANGE LEVEL PROT I
1750	VCS-ILT00495-MS	2	10/09/89	INST LOOP DIAG - SGC NARROW RANGE LEVEL PROT II
1751	VCS-ILT00496-MS	2	10/09/89	INST LOOP DIAG - SGC NARROW RANGE LEVEL PROT III
1753	VCS-ILT00920-SI	2	09/11/90	INST LOOP DIAG - SI ACCUMULATOR A LEVEL
1754	VCS-ILT00922-SI	1	05/25/88	INST LOOP DIAG - SI ACCUMULATOR A LEVEL
1755	VCS-ILT00924-SI	1	05/25/88	INST LOOP DIAG - SI ACCUMULATOR B LEVEL
1756	VCS-ILT00926-SI	1	05/25/88	INST LOOP DIAG - SI ACCUMULATOR B LEVEL
1757	VCS-ILT00928-SI	1	05/25/88	INST LOOP DIAG - SI ACCUMULATOR C LEVEL
1758	VCS-ILT00930-SI	0	01/18/89	INST LOOP DIAG - SI ACCUMULATOR C LEVEL
1759	VCS-ILT00990-SF	1	10/09/89	INST LOOP DIAG - RWST LEVEL
1760	VCS-ILT00991-SF	1	10/09/89	INST LOOP DIAG - RWST LEVEL # 2
1761	VCS-ILT00992-SF	0	02/26/90	INST LOOP DIAG - RWST LEVEL
1762	VCS-ILT00993-SF	1	10/09/89	INST LOOP DIAG - RWST LEVEL # 4
1951	VCS-ILT01310-RC	0	11/07/89	INST LOOP DIAG - REACTOR VESSEL UPPER PLENUM LEVEL (TRAIN A)
1952	VCS-ILT01311-RC	0	02/28/90	INST LOOP DIAG - REACTOR VESSEL NARROW RANGE WATER LEVEL (TRAIN A)
1953	VCS-ILT01320-RC	0	02/28/90	INST LOOP DIAG - REACTOR VESSEL UPPER PLENUM LEVEL (TRAIN B)
1954	VCS-ILT01321-RC	0	02/22/90	INST LOOP DIAG - REACTOR VESSEL NARROW RANGE LEVEL (TRAIN B)
1955	VCS-ILT01322-RC	0	02/28/90	INST LOOP DIAG - WIDE RANGE REACTOR VESSEL LEVEL (TRAIN B)
1763	VCS-ILT01963-LD	0	11/09/88	INST LOOP DIAG - RB SUMP LEVEL (TRAIN A)

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1764	VCS-ILT01964-LD	0	01/03/89	INST LOOP DIAG - RB SUMP LEVEL (TRAIN B)
1956	VCS-ILT01969-LD	0	09/28/87	INST LOOP DIAG - RB RHR SUMP A LEVEL
1957	VCS-ILT01970-LD	0	11/09/88	INST LOOP DIAG - RB RHR SUMP B LEVEL
1958	VCS-ILT03010-CO	0	02/27/90	INST LOOP DIAG - CONDENSER B HOTWELL LEVEL
1959	VCS-ILT03135-CO	1	12/18/90	INST LOOP DIAG - DEAER STORAGE TANK WIDE RANGE LEVEL
1960	VCS-ILT03136-CO	0	12/18/90	INST LOOP DIAG - DEAER STORAGE TANK LEVEL
1765	VCS-ILT03142-CO	0	05/25/89	INST LOOP DIAG - FW TURBINE A CONDENSER HOTWE[illegible]VEL
1766	VCS-ILT03152-CO	0	02/14/90	INST LOOP DIAG - FW TURBINE B CONDENSER HOTWEL[illegible]EVEL
1767	VCS-ILT03162-CO	0	02/14/90	INST LOOP DIAG - FW TURBINE C CONDENSER HOTWELL LEVEL
1730	VCS-ILT03621-EF	0	03/22/90	INST LOOP DIAG - CONDENSATE STORAGE TANK LEVEL (TRAIN A)
1731	VCS-ILT03631-EF	0	03/22/90	INST LOOP DIAG - CONDENSATE STORAGE TANK LEVEL (TRAIN B)
1961	VCS-ILT04003-CW	0	03/05/90	INST LOOP DIAG - LAKE MONTICELLO WATER LEVEL
1768	VCS-ILT04413-SW	0	02/25/88	INST LOOP DIAG - SW POND LEVEL (TRAIN A)
1769	VCS-ILT04458-SW	0	09/01/89	INST LOOP DIAG - SW POND LEVEL (TRAIN B)
1770	VCS-ILT07092-CC	0	04/20/89	INST LOOP DIAG - CCW SURGE TANK LEVEL (TRAIN B)
1771	VCS-ILT07094-CC	0	05/15/89	INST LOOP DIAG - CCW SURGE TANK LEVEL
1772	VCS-ILT07356-SP	0	06/14/88	INST LOOP DIAG - SODIUM HYDROXIDE STORAGE TANK LEVEL (TRAIN A)
1773	VCS-ILT07358-SP	0	06/14/88	INST LOOP DIAG - SODIUM HYDROXIDE STORAGE TANK LEVEL (TRAIN B)
1962	VCS-ILT07403-SF	0	11/23/88	INST LOOP DIAG - SF REFUEL CAVITY LEVEL
1963	VCS-ILT07405-SF	0	07/22/88	INST LOOP DIAG - SF TRANSFER CANAL LEVEL
1964	VCS-ILT07407-SF	0	06/21/89	INST LOOP DIAG - SF POOL WATER LEVEL (TRAIN A)
1965	VCS-ILT07431-SF	0	06/21/89	INST LOOP DIAG - SF POOL WATER LEVEL (TRAIN A)
1966	VCS-ILT07433-SF	0	02/22/90	INST LOOP DIAG - SF POOL WATER LEVEL (TRAIN B)
1774	VCS-ILT07603-MU	0	12/01/88	INST LOOP DIAG - RMU STORAGE TANK LEVEL
1775	VCS-ILT07604-MU	0	01/18/89	INST LOOP DIAG - RMU WATER STORAGE TANK LEVEL (B LOOP)
1862	VCS-INE00031-NI	0	03/05/90	INST LOOP DIAG - NIS SOURCE RANGE NEUTRON FLUX CHANNEL II

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1863	VCS-INE00032-NI	0	03/05/90	INST LOOP DIAG - NIS SOURCE RANGE NEUTRON FLUX CHANNEL II
1864	VCS-INE00035-NI	0	03/13/90	INST LOOP DIAG - IR NEUT LEVEL
1865	VCS-INE00036-NI	0	11/03/89	INST LOOP DIAG - NIS INTERMEDIATE RANGE NEUTRON FLUX CHANNEL II
1866	VCS-INE00041-NI	0	03/22/90	INST LOOP DIAG - NIS POWER
1867	VCS-INE00042-NI	0	03/16/90	INST LOOP DIAG - NIS POWER RANGE NEUTRON FLUX CHANNEL II
1868	VCS-INE00043-NI	0	03/19/90	INST LOOP DIAG - NIS POWER
1869	VCS-INE00044-NI	0	03/26/90	INST LOOP DIAG - NIS POWER RANGE NEUTRON FLUX CHANNEL IV
1870	VCS-INY00037/36-NI	0	10/30/89	INST LOOP DIAG - NIS COMPARATOR & RATE INSTRUMENTATION
1871	VCS-INY00050-NI	0	10/30/89	INST LOOP DIAG - NIS FLUX DEVIATION & MISC INDICATION & CONTROL
1776	VCS-IPT00117-CS	1	05/25/88	INST LOOP DIAG - VOLUME CONTROL TANK PRESSURE
1967	VCS-IPT00121-CS	0	12/14/86	INST LOOP DIAG - CHARGING HEADER PRESSURE
1968	VCS-IPT00138-CS	1	06/18/90	INST LOOP DIAG - EXCESS LETDOWN HX OUTLET PRESSURE
1777	VCS-IPT00145-CS	1	06/18/90	INST LOOP DIAG - LOW PRESSURE LETDOWN PRESSURE
1778	VCS-IPT00154-CS	3	10/06/89	INST LOOP DIAG - RCP C # 1 SEAL DIFFERENTIAL PRESSURE
1779	VCS-IPT00155-CS	2	10/06/89	INST LOOP DIAG - RCP B # 1 SEAL DIFFERENTIAL PRESSURE
1780	VCS-IPT00156-CS	2	06/12/90	INST LOOP DIAG - RCP A # 1 SEAL DIFFERENTIAL PRESSURE
1781	VCS-IPT00402-RC	2	06/12/90	INST LOOP DIAG - RC LOOP 3 HOT LEG PRESSURE
1782	VCS-IPT00403-RC	2	06/12/90	INST LOOP DIAG - RC LOOP 1 HOT LEG PRESSURE
1783	VCS-IPT00444-RC	2	06/18/90	INST LOOP DIAG - PRZR PRESSURE
1784	VCS-IPT00445-RC	1	05/25/88	INST LOOP DIAG - PRZR PRESSURE
1785	VCS-IPT00446-RC	0	08/29/89	INST LOOP DIAG - FIRST STAGE TURBINE PRESSURE
1786	VCS-IPT00455-RC	1	10/09/89	INST LOOP DIAG - PRZR PRESSURE
1787	VCS-IPT00456-RC	1	10/09/89	INST LOOP DIAG - PRZR PRESSURE
1788	VCS-IPT00457-RC	3	10/09/89	INST LOOP DIAG - PRZR PRESSURE
1972	VCS-IPT00464-RC	0	06/10/87	INST LOOP DIAG - MAIN STEAM HEADER PRESSURE

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1789	VCS-IPT00472-RC	0	08/17/87	INST LOOP DIAG - PRZR RELIEF TANK PRESSURE
1790	VCS-IPT00474-MS	2	10/09/89	INST LOOP DIAG - SGA MAIN STEAM HEADER PRESSURE
1791	VCS-IPT00475-MS	3	10/09/89	INST LOOP DIAG - SGA MAIN STEAM HEADER PRESSURE
1792	VCS-IPT00476-MS	2	10/09/89	INST LOOP DIAG - SGA MAIN STEAM HEADER PRESSURE
1793	VCS-IPT00484-MS	2	10/09/89	INST LOOP DIAG - SGB MAIN STEAM HEADER PRESSURE
1794	VCS-IPT00485-MS	1	10/06/89	INST LOOP DIAG - SGB MAIN STEAM HEADER PRESSURE
1795	VCS-IPT00486-MS	1	10/09/89	INST LOOP DIAG - SGB MAIN STEAM HEADER PRESSURE
1796	VCS-IPT00494-MS	2	10/09/89	INST LOOP DIAG - SGC MAIN STEAM HEADER PRESSURE
1797	VCS-IPT00495-MS	1	10/09/89	INST LOOP DIAG - SGC MAIN STEAM HEADER PRESSURE
1798	VCS-IPT00496-MS	3	10/09/89	INST LOOP DIAG - SGC MAIN STEAM HEADER PRESSURE
1973	VCS-IPT00508-FW	1	10/23/90	INST LOOP DIAG - FWP SPEED CONTROL
1799	VCS-IPT00600A-RH	2	09/11/90	INST LOOP DIAG - RHR PMP A DISCHARGE PRESSURE
1800	VCS-IPT00600B-RH	1	09/11/90	INST LOOP DIAG - RHR PMP B DISCHARGE PRESSURE
1801	VCS-IPT00921-SI	3	09/11/90	INST LOOP DIAG - SI ACCUMULATOR A PRESSURE
1802	VCS-IPT00923-SI	2	08/21/89	INST LOOP DIAG - SI ACCUMULATOR A PRESSURE
1803	VCS-IPT00925-SI	2	08/21/89	INST LOOP DIAG - SI ACCUMULATOR B PRESSURE
1804	VCS-IPT00927-SI	3	08/01/90	INST LOOP DIAG - SI ACCUMULATOR B PRESSURE
1805	VCS-IPT00929-SI	2	02/21/89	INST LOOP DIAG - SI ACCUMULATOR C PRESSURE
1806	VCS-IPT00931-SI	2	08/21/89	INST LOOP DIAG - SI ACCUMULATOR C PRESSURE
1807	VCS-IPT00950-SI	0	01/03/89	INST LOOP DIAG - CONTAINMENT PRESSURE PROTECTION CHANNEL I
1808	VCS-IPT00951-SI	1	10/09/89	INST LOOP DIAG - CONTAINMENT PRESSURE PROTECTION CHANNEL II
1809	VCS-IPT00952-SI	1	10/09/89	INST LOOP DIAG - CONTAINMENT PRESSURE PROTECTION CHANNEL III
1810	VCS-IPT00953-SI	1	10/09/89	INST LOOP DIAG - CONTAINMENT PRESSURE PROTECTION CHANNEL IV
1974	VCS-IPT00954A-HR	1	10/09/89	INST LOOP DIAG - RB WIDE RANGE PRESSURE
1975	VCS-IPT00954B-HR	0	01/30/89	INST LOOP DIAG - RB WIDE RANGE PRESSURE
1811	VCS-IPT02000-MS	0	09/12/89	INST LOOP DIAG - SGA OUTLET PRESSURE

DOC NUM	DOC NAME	REV	DATE	TITLE
1812	VCS-IPT02010-MS	0	10/16/89	INST LOOP DIAG - SGB OUTLET PRESSURE
1813	VCS-IPT02020-MS	0	02/22/90	INST LOOP DIAG - SGC OUTLET PRESSURE
1976	VCS-IPT02032-MS	0	11/17/88	INST LOOP DIAG - MAIN STEAM TO EFW TURBINE PRESSURE
1977	VCS-IPT02074-MS	0	11/23/88	INST LOOP DIAG - MOIST SEPARATOR A PRESSURE
1978	VCS-IPT02084-MS	0	11/23/88	INST LOOP DIAG - MOIST SEPARATOR B PRESSURE
1814	VCS-IPT03006-CO	0	07/14/88	INST LOOP DIAG - CONDENSER A HOTWELL PRESSURE
1815	VCS-IPT03016-CO	0	07/14/88	INST LOOP DIAG - CONDENSER B HOTWELL PRESSURE
1979	VCS-IPT03140-CO	0	10/20/88	INST LOOP DIAG - FW TURBINE A CONDENSER PRESSURE
1980	VCS-IPT03150-CO	0	10/09/89	INST LOOP DIAG - FW TURBINE B CONDENSER PRESSURE
1981	VCS-IPT03160-CO	1	10/09/89	INST LOOP DIAG - FW TURBINE C CONDENSER PRESSURE
1982	VCS-IPT03211-FW	0	12/01/88	INST LOOP DIAG - DEAER SHELL PRESSURE
1816	VCS-IPT03233-FW	0	12/21/88	INST LOOP DIAG - FW BOOSTER PMP DISCHARGE HEADER PRESSURE
1983	VCS-IPT03243-FW	0	03/17/88	INST LOOP DIAG - MAIN FWP A DISCHARGE PRESSURE
1984	VCS-IPT03253-FW	0	10/20/88	INST LOOP DIAG - MAIN FWP B DISCHARGE PRESSURE
1985	VCS-IPT03263-FW	0	03/31/88	INST LOOP DIAG - MAIN FWP C DISCHARGE PRESSURE
1986	VCS-IPT03563-EF	0	07/11/88	INST LOOP DIAG - EFW TO SGA PRESSURE
1987	VCS-IPT03573-EF	0	07/28/88	INST LOOP DIAG - EFW TO SGB PRESSURE
1988	VCS-IPT03583-EF	0	11/17/88	INST LOOP DIAG - EFW TO SGC PRESSURE
1989	VCS-IPT03632-EF	0	02/14/90	INST LOOP DIAG - EMERG AUX FW SUCTION PRESSURE
1817	VCS-IPT03633-EF	0	02/14/90	INST LOOP DIAG - EMERG AUX FW SUCTION PRESSURE
1990	VCS-IPT03634-EF	0	03/22/90	INST LOOP DIAG - EMERG AUX FW SUCTION PRESSURE
1991	VCS-IPT04036-CW	0	08/26/88	INST LOOP DIAG - CW DISCHARGE HEADER PRESSURE
1992	VCS-IPT04152-CW	0	08/01/88	INST LOOP DIAG - CW OPEN CYCLE BOOSTER PMP A DISCHARGE PRESSURE
1993	VCS-IPT04162-CW	0	10/13/88	INST LOOP DIAG - CW OPEN CYCLE BOOSTER PMP B DISCHARGE PRESSURE
1818	VCS-IPT04402-SW	0	02/14/90	INST LOOP DIAG - SWP A DISCHARGE PRESSURE

DOC NUM	DOC NAME	REV	DATE	TITLE
1819	VCS-IPT04422-SW	0	06/28/89	INST LOOP DIAG - SWP B DISCHARGE PRESSURE
1994	VCS-IPT04442-SW	0	02/19/90	INST LOOP DIAG - SERVICE WATER PUMP C DISCHARGE PRESSURE
1820	VCS-IPT04443-SW	0	02/22/90	INST LOOP DIAG - SWP C DISCHARGE PRESSURE
1821	VCS-IPT04523-SW	0	01/03/89	INST LOOP DIAG - SW BOOSTER PMP A DISCHARGE PRESSURE
1822	VCS-IPT04528-SW	0	02/22/90	INST LOOP DIAG - SW FROM RBCU A HEADER PRESSURE
1823	VCS-IPT04543-SW	0	11/17/88	INST LOOP DIAG - SW BOOSTER PMP B DISCHARGE PRESSURE
1824	VCS-IPT04548-SW	0	01/03/89	INST LOOP DIAG - SW BOOSTER PMP B DISCHARGE PRESSURE
1825	VCS-IPT07032-CC	0	01/18/89	INST LOOP DIAG - CC DISCHARGE HEADER A PRESSURE
1826	VCS-IPT07042-CC	0	11/17/89	INST LOOP DIAG - CC DISCHARGE HEADER B PRESSURE
1995	VCS-IPT07367-SP	0	06/28/88	INST LOOP DIAG - RB SPRAY PMP A DISCHARGE PRESSURE
1996	VCS-IPT07377-SP	0	06/28/88	INST LOOP DIAG - RB SPRAY PMP B DISCHARGE PRESSURE
1997	VCS-IPT07603-MU	0	10/20/88	INST LOOP DIAG - RMU WATER PMP A DISCHARGE PRESSURE
1998	VCS-IPT07612-MU	0	03/17/88	INST LOOP DIAG - DEGAS DEMIN WATER TO RMU STORAGE TANK PRESSURE
1999	VCS-IPT07618-MU	0	01/18/89	INST LOOP DIAG - RMU WATER PMP B DISCHARGE PRESSURE
1827	VCS-IPT08254-RF	0	01/18/89	INST LOOP DIAG - RB NARROW RANGE PRESSURE
2000	VCS-IPT08342-IA	0	08/22/88	INST LOOP DIAG - IA HEADER PRESSURE
2001	VCS-IPT08386-IA	0	03/23/88	INST LOOP DIAG - RB IA SUPPLY PRESSURE
1828	VCS-ITE00116-CS	0	03/17/86	INST LOOP DIAG - VOLUME CONTROL TANK TEMP
2002	VCS-ITE00123-CS	0	08/29/86	INST LOOP DIAG - REGEN HX OUTLET TEMPERATURE
1829	VCS-ITE00139-CS	0	07/10/86	INST LOOP DIAG - EXCESS LETDOWN HX OUTLET TEMPERATURE
1830	VCS-ITE00140-CS	0	07/10/86	INST LOOP DIAG - REGEN HX LETDOWN OUTLET TEMPERATURE
1831	VCS-ITE00141-CS	0	07/14/86	INST LOOP DIAG - LETDOWN HEADER RELIEF OUTLET TEMPERATURE
1832	VCS-ITE00143-CS	0	06/18/83	INST LOOP DIAG - DEMIN FLOW DIVERT RTD
2003	VCS-ITE00144-CS	0	07/08/87	INST LOOP DIAG - LETDOWN HX OUTLET TEMPERATURE
2004	VCS-ITE00376-TR	0	06/28/86	INST LOOP DIAG - BTRS CHILLER OUTLET TEMPERATURE
2005	VCS-ITE00379-TR	0	07/18/88	INST LOOP DIAG - BTRS CHILLER SURGE TANK TEMPERATURE

DOC NUM	DOC NAME	REV	DATE	TITLE
1833	VCS-ITE00381-TR	0	01/18/89	INST LOOP DIAG - LETDOWN HX OUTLET TEMPERATURE
1834	VCS-ITE00386-TR	1	09/11/90	INST LOOP DIAG - LETDOWN CHILLER HX OUTLET TEMPERATURE
2006	VCS-ITE00389-TR	0	04/28/88	INST LOOP DIAG - THERMAL REGEN DEMIN OUTLET TEMPERATURE
1835	VCS-ITE00401-RC	0	09/03/86	INST LOOP DIAG - REACTOR VESSEL FLANGE LEAKOFF TEMPERATURE
1836	VCS-ITE00410-RC	2	09/11/90	INST LOOP DIAG - RC LOOP A WIDE RANGE COLD LEG TEMPERATURE
2007	VCS-ITE00412-RC	0	02/27/89	INST LOOP DIAG - RCL A HOT LEG/COLD LEG NR RTD
1837	VCS-ITE00413-RC	2	09/11/90	INST LOOP DIAG - RC LOOP A WIDE RANGE HOT LEG TEMPERATURE
2008	VCS-ITE00420-RC	1	08/02/89	INST LOOP DIAG - RC LOOP B WIDE RANGE COLD LEG TEMPERATURE
2009	VCS-ITE00422-RC	0	02/27/89	INST LOOP DIAG - RCL B HOT LEG/COLD LEG NR RTD
2010	VCS-ITE00423-RC	1	08/02/89	INST LOOP DIAG - RC LOOP B WIDE RANGE HOT LEG TEMPERATURE
2011	VCS-ITE00432-RC	0	02/23/89	INST LOOP DIAG - RCL C HOT LEG/COLD LEG NR RTD
2012	VCS-ITE00433-RC	1	08/02/89	INST LOOP DIAG - RC LOOP C WIDE RANGE HOT LEG TEMPERATURE
1838	VCS-ITE00450-RC	0	07/14/88	INST LOOP DIAG - PRZR SURGE LINE TEMPERATURE
1839	VCS-ITE00451-RC	0	09/12/86	INST LOOP DIAG - PRZR SPRAY LINE TEMPERATURE
1840	VCS-ITE00452-RC	0	09/28/87	INST LOOP DIAG - PRZR SPRAY LINE TEMPERATURE
1841	VCS-ITE00453-RC	0	07/18/88	INST LOOP DIAG - PRZR WATER TEMPERATURE
1842	VCS-ITE00454-RC	0	02/18/88	INST LOOP DIAG - PRZR VAPOR TEMPERATURE
1843	VCS-ITE00463-RC	0	08/31/87	INST LOOP DIAG - PRZR PWR RELIEF LINE TEMPERATURE
1844	VCS-ITE00465-RC	0	03/14/88	INST LOOP DIAG - PRZR SAFETY VALVE LINE TEMPERATURE
1845	VCS-ITE00467-RC	0	03/14/88	INST LOOP DIAG - PRZR SAFETY VALVE LINE TEMPERATURE
1846	VCS-ITE00469-RC	0	03/14/88	INST LOOP DIAG - PRZR SAFETY VALVE LINE TEMPERATURE
1847	VCS-ITE00471-RC	0	08/18/87	INST LOOP DIAG - PRZR RELIEF TANK WATER TEMPERATURE
2013	VCS-ITE00604A-RH	0	09/05/86	INST LOOP DIAG - RHR PMP 1 DISCHARGE TEMPERATURE
2014	VCS-ITE00604B-RH	0	04/24/86	INST LOOP DIAG - RHR PMP 2 DISCHARGE TEMPERATURE
2015	VCS-ITE00606A-RH	1	05/28/88	INST LOOP DIAG - RHR HX XHE5A-RH OUTLET TEMPERATURE

DOC NUM	DOC NAME	REV	DATE	TITLE
2016	VCS-ITE00606B-RH	1	0[illegible]/[illegible]8/88	INST LOOP DIAG - RHR HX XHE5B-RH OUTLET TEMPERATURE
2017	VCS-ITE01971-LD	0	[illegible]/09/88	INST LOOP DIAG - RB SUMP TEMPERATURE (TRAIN A)
2018	VCS-ITE01972-LD	0	10/20/88	INST LOOP DIAG - RB SUMP TEMPERATURE (TRAIN B)
2019	VCS-ITE03222-FW	2	12/18/90	INST LOOP DIAG - DEAER STORAGE TANK TEMPERAT[illegible]
1883	VCS-ITE03307A-FW	0	05/15/89	INST LOOP DIAG - FEEDWATER HEADER TRAD
1884	VCS-ITE03307B-FW	0	05/15/89	INST LOOP DIAG - FEEDWATER HEADER TRBD
1848	VCS-ITE03318-FW	0	08/05/88	INST LOOP DIAG - FW TO SGA TEMPERATURE (TRAIN B)
1849	VCS-ITE03322-FW	0	08/05/88	INST LOOP DIAG - FW TO SGA TEMPERATURE (TRAIN A)
1850	VCS-ITE03328-FW	0	08/05/88	INST LOOP DIAG - FW TO SGB TEMPERATURE (TRAIN B)
1851	VCS-ITE03332-FW	0	01/05/89	INST LOOP DIAG - FW TO SGB TEMPERATURE (TRAIN A)
1852	VCS-ITE03338-FW	0	08/05/88	INST LOOP DIAG - FW TO SGC TEMPERATURE (TRAIN B)
1853	VCS-ITE03342-FW	0	01/05/89	INST LOOP DIAG - FW TO SGC TEMPERATURE (TRAIN A)
2020	VCS-ITE04035-CW	0	08/26/88	INST LOOP DIAG - CW DISCHARGE HEADER TEMPERATURE
2021	VCS-ITE04211-CW	0	09/13/89	INST LOOP DIAG - TURBINE OIL RESERVOIR TEMPERATURE
1854	VCS-ITE04265-CW	0	02/22/90	INST LOOP DIAG - GENERATOR H2 GAS TEMPERATURE
1856	VCS-ITE04270-CW	0	06/29/89	INST LOOP DIAG - ALTERNATOR COOLER TEMPERATURE
2022	VCS-ITE04467-SW	0	07/07/88	INST LOOP DIAG - SW LOOP A OUTLET TEMPERATURE
2023	VCS-ITE04480-SW	0	06/29/88	INST LOOP DIAG - SW LOOP A INLET TEMPERATURE
2024	VCS-ITE04497-SW	0	07/26/88	INST LOOP DIAG - SW LOOP B OUTLET TEMPERATURE
2025	VCS-ITE04510-SW	0	07/06/88	INST LOOP DIAG - SW LOOP B INLET TEMPERATURE
1857	VCS-ITE04606-TC	0	03/14/89	INST LOOP DIAG - TURBINE RM CLOSED CYCLE CODING HX DISCHARGE HEADER TEMPERATUR
1858	VCS-ITE07108-CC	0	05/16/89	INST LOOP DIAG - EXCESS LETDOWN HX CC OUTLET TEMPERATURE
2026	VCS-ITE07401-SF	0	11/12/87	INST LOOP DIAG - SF REFUELING CAVITY TEMPERATURE
2027	VCS-ITE07435-SF	0	02/06/89	INST LOOP DIAG - SF POOL TEMPERATURE
2028	VCS-ITE07437-SF	0	06/29/88	INST LOOP DIAG - SF POOL TEMPERATURE (B LOOP)
2029	VCS-ITE07505-RW	0	12/01/88	INST LOOP DIAG - RWST TEMPERATURE (A LOOP)

DOC NUM	DOC NAME	REV	DATE	TITLE
2030	VCS-ITE07507-RW	0	10/12/98	INST LOOP DIAG - RWST TEMPERATURE
1859	VCS-ITE07601-MU	0	01/18/89	INST LOOP D - RMU STORAGE TANK TEMPERATURE
1860	VCS-ITE07602-MU	0	12/01/88	INST LOOP DIAG RMU STORAGE TANK TEMPERATURE
2031	VCS-ITE09013-VU	0	12/01/88	INST LOOP DIAG - CHILLER A OUTLET TEMPERATURE
2032	VCS-ITE09023-VU	0	01/31/89	INST LOOP DIAG - CHILLER B OUTLET TEMPERATURE
2033	VCS-ITE09033A-VU	0	06/15/89	INST LOOP DIAG - CHILLER C OUTLET TEMPERATURE
2034	VCS-ITE09033B-VU	0	01/18/89	INST LOOP DIAG - CHILLER C OUTLET TEMPERATURE (TRAIN B)
2035	VCS-ITE09203A-CI	0	09/13/89	INST LOOP DIAG - RB TEMPERATURE
2036	VCS-ITE09205-CI	0	03/14/90	INST LOOP DIAG - RB TEMPERATURE
1861	VCS-ITE09213-CI	0	06/15/89	INST LOOP DIAG - IC SUPPLY HEADER HI TEMPERATURE
1855	VCS-ITE09341-AH	0	01/03/89	INST LOOP DIAG - CRDM SHROUD EXIT TEMPERATURE
2038	VCS-ITE09961-AH	0	11/17/88	INST LOOP DIAG - SW PMP RM AMBIENT TEMPERATURE
2039	VCS-ITE09962-AH	0	11/17/88	INST LOOP DIAG - SW PMP & SCREEN RM AREA TEMPERATURE
2040	VCS-ITE09963-AH	0	11/23/88	INST LOOP DIAG - SW BLDG SWITCHGEAR RM IEA AMBIENT TEMPERATURE
2041	VCS-ITE09964-AH	0	11/17/88	INST LOOP DIAG - SW BLDG 7200 VOLT SWITCHGEAR RM TEMPERATURE
2042	VCS-ITE09965-AH	0	01/30/89	INST LOOP DIAG - SW BLDG MOTOR CONTROL RM AMBIENT TEMPERATURE
2043	VCS-ITE09966-AH	0	10/25/88	INST LOOP DIAG - SW BLDG MOTOR CONTROL CENTER RM TEMPERATURE
1872	VCS-ITY00408-RC	1	01/11/91	INST LOOP DIAG - MS DUMP SYS & ROD SPEED CONTROL
1873	VCS-ITY00409-RC	1	09/11/90	INST LOOP DIAG - DELTA T AUCTIONEERED CONTROL & ALARM
1874	VCS-IZY00409A-RC	0	03/08/89	INST LOOP DIAG - CONTROL ROD BANK A INSERTION LIMITS
1875	VCS-IZY00409B-RC	0	01/06/89	INST LOOP DIAG - CONTROL ROD BANK B INSERTION LIMITS
1876	VCS-IZY00409C-RC	0	01/C5/89	INST LOOP DIAG - CONTROL ROD BANK C INSERTION LIMITS
1877	VCS-IZY00409D-RC	0	01/03/89	INST LOOP DIAG - CONTROL ROD BANK D INSERTION LIMITS
1608	WESTINGHOUSE		12/26/90	MESSAGE FROM MEKUND BORLE
1599	XVT02843		**/**/**	VALVE TAG

DOC NUM	DOC NAME	REV	DATE	TITLE

Total Source Documents Referenced in CMS: 916

SIMULATOR DESIGN BASIS DOCUMENTATION

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DESIGN BASIS AND MODEL DOCUMENTATION
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SIMULATOR TEST CERTIFICATION PANEL

A. PURPOSE

To ensure that the response of each malfunction or transient modeled for the Virgil C. Summer Simulator are the responses that would be expected to occur if that malfunction or transient were to occur at the Virgil C. Summer Nuclear Station, and best estimate criteria must be utilized.

B. PANEL AND QUALIFICATIONS

1. Simulator Operations Specialist, SRO/SRC, Chairman
2. Training Instructor, SRO/SRC
3. Operations, SRO
4. Engineering, SRO/SROC
5. Simulator Operations Specialist, Westinghouse

C. EXCEPTIONS

This procedure is not applicable for malfunctions and transients for which actual plant results or plant design data are available. This exception excludes FSAR worst case situations since FSAR data are available for these events.

D. RESPONSIBILITIES

1. A Simulator Operations Specialist chairs the Simulator Test Certification Panel.

2. The Simulator Test Certification Panel is responsible for recommending the responses contained in each malfunction or transient test to be certified.

E. INSTRUCTIONS

1. Each malfunction or transient test response shall be reviewed by the Simulator Test Certification Panel.

2. The malfunction or transient response is recommended by a three-fifths (3/5), simple majority poll. Every effort shall be made to resolve any differences. However, where differences remain, the dissenting

opinion(s) shall be maintained with the approved original of the malfunction or transient test procedure.

3. The malfunction or transient response recommended for certification, along with any differing professional opinions, are submitted in a package to the Supervisor, Simulator and Training Development, for final approval. The malfunction or transient test may be approved or returned to the Simulator Test Certification Panel for further development.

4. The approved, original malfunction or transient test procedure and any differing professional opinions are maintained in a separate, certified malfunction/transient test procedure file.

5. Minor changes may be made to the malfunction or transient test procedure as a result of testing to account for minor differences between the simulated response and the expected response. Differences which cannot be characterized as minor require the review of the Simulator Test Certification Panel.

3.0 SIMULATOR TESTS

3.0.1 Conduct of Testing Operations

IST-1.0 (Appendix 1 to Section 3.0.1) describes the certification testing plan for the Virgil C. Summer Nuclear Station Simulator. It summarizes the testing to be done, acceptance criteria and discrepancy resolution method used in demonstrating conformance to the testing requirements of ANSI/ANS 3.5-1985 and Regulatory Guide 1.149-1985. Please note that the four-year recertification testing plan has been provided as Appendix I to Section 3.5.

NUCLEAR OPERATIONS EDUCATION

AND TRAINING

V.C. SUMMER TRAINING SIMULATOR
INTEGRATED SYSTEMS TEST
CONDUCT OF OPERATIONS

IST-1.0
REVISION 8
03/18/91

RECOMMENDED BY Frank Blanchard DATE 3/25/91
SIMULATOR OPERATIONS SPECIALIST

APPROVED Lindsay J. Bly For Peter L. Walker DATE 3/25/91
WESTINGHOUSE CERTIFICATION COORDINATOR

APPROVED DATE 3/25/91
SUPERVISOR, SIMULATOR AND TRAINING DEVELOPMENT

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OBJECTIVES

This document details the integrated systems testing plan for the V.C. Summer Nuclear Station Training Simulator. It contains the necessary information to inform those persons conducting tests on the simulator as to the methods, procedures and practices to be used to systematically conduct prescribed tests, collect pertinent information and document test results. These tests are organized to ensure the simulator does, in fact, realistically simulate the V.C. Summer Nuclear Station and meet the requirements of ANSI/ANS 3.5-1985, and Regulatory Guide 1.149-1985.

1.0 TYPES OF INTEGRATED TESTS

The Integrated Systems Test Plan is organized into nine (9) main areas: Normal Operations Tests, Initial Condition Checks, Steady State Heat Balance Tests, Startup Tests, Malfunction Tests, Transient Tests, Hardware Tests, Plant LER Tests and Real Time Tests.

1.1 Initial Condition Checks

The objectives of these tests are to verify the simulator initial conditions correspond to typical V.C. Summer Nuclear Station operating conditions and the initial conditions are stable. Simulator initial conditions are selected for their training value and represent a variety of operating conditions, fission product poison concentrations and core ages. The initial conditions simulated represent major plant evolutions; i.e., Mode 3 going to Mode 4, or critical equipment operation; i.e., startup and synchronization of the main generator. The V.C. Summer Nuclear Station Technical Specifications and Operating Procedures are used as the reference data for the simulator initial conditions. The simulator initial conditions will be compared with the corresponding reference plant conditions as specified in the V.C. Summer Nuclear Station General Operating Procedures (GOPs).

1.1.1 Acceptance Criteria for Initial Conditions Checks

Acceptance criteria for initial conditions checks are as follows:

a. The initial condition is stable: no immediate operator manual action is required to maintain the desired plant condition.

b. The initial condition systems alignment, equipment in operation, and pressure/temperature conditions satisfy the conditions specified in the referenced V.C. Summer Nuclear Station General Operating Procedure.

1.2 Steady State Accuracy and Heat Balance Tests

The objectives of these tests are to demonstrate the proper relationship among the plant process variables at various plant steady state conditions and compatibility with actual V.C. Summer Nuclear Station values. The reference data used for comparison will include V.C. Summer Nuclear Station heat balances, mass balances, plant computer logs, plant computer print value reviews, surveillance test procedures, and any other available reference data. The simulator will be in the run mode for at least one (1) hour while the various comparisons are being made to insure that the simulation is truly at a steady-state condition. Three initial conditions have been chosen to represent the full range of power operations, all at middle of core life conditions. Middle of core life is chosen for most of the initial conditions since this is the most representative of all core ages for operator training. Testing at the following power levels is performed annually; all other initial conditions at these power levels will be checked during a 4-year period.

100% power: This is the normal mode of plant operations.

75% power: This condition is convenient for heat balances and mass calculations for the primary and secondary plant.

50% power: Same comment as for 75% power.

1.2.1 Acceptance Criteria for Steady-State Accuracy and Heat Balances

Acceptance criteria for steady state accuracy and heat balances are as follows:

a. Shall be stable and not vary more than ± 2% of the initial values over 60 minutes.

b. Critical Parameters shall agree within ± 2% plus instrument tolerances of the V.C. Summer Nuclear Station and not detract from training.

c. Non-Critical Parameters pertinent to operations shall agree within ± 10% of the V.C. Summer Nuclear Station data.

d. Critical and Non-Critical Parameters pertinent to operations shall be accurate to the extent as to not detract from training.

e. Mass and energy balances shall be accurate to ± 2%, or ± 10% as specified on the appropriate attachment sheets.

f. Instrument error shall be no greater than the allowances specified on the V.C. Summer Nuclear Station control room logs.

1.3 Startup Testing

Initial criticality testing based on the V.C. Summer Nuclear Station simulator automatic test program entitled "TESTRX PROGRAM", written especially for the V.C. Summer Simulator will be utilized to determine the following:

a. Determine the Isothermal Temperature Coefficient of Reactivity. The Isothermal Temperature Coefficient (TISO) is defined as the change in core reactivity per unit change in moderator, clad and fuel temperature.

b. Determine Boron Worth.

c. Determine the Reactivity Worth of the RCC Banks.

d. Determine the simulators ability to achieve criticality within ± 50 steps (± 500 pcm) of the V.C. Summer Nuclear Station core through all simulated core ages and xenon or samarium concentrations.

1.3.1 Acceptance Criteria For Startup Testing

The acceptance criteria for startup testing are based on the appropriate Westinghouse Nuclear Design and Operations Package (WCAP) for the V.C. Summer Nuclear Station and are as follows:

a. Control Rod Bank Integral Worth is within ± 10% of the design predicted integral reactivity worth.

b. The Differential Boron Reactivity Worth is within ± 10% of the design predicted boron worth.

c. The Isothermal Temperature Coefficient of Reactivity is within ± 3 pcm/°F of the design predicted temperature coefficient.

1.4 Normal Operations Testing

The simulator will be operated over the full range of plant operations from full power to cold shutdown and back according to the following schedule:

a. Power Reduction to 75 Percent Power.

(Includes 10% step load decrease).

b. Power Reduction to 50 Percent Power.

c. Power Reduction to 2 Percent Power.

d. 2 Percent Power to Hot Standby.

e. Hot Standby to Hot Shutdown.

f. Hot Shutdown to Cold Shutdown.

g. Cold Shutdown to Hot Shutdown.

h. Hot Shutdown to Hot Standby.

i. Hot Standby to 2 Percent Power.

j. 2 Percent Power to 100 Percent Power.

k. Recovery to Rated Power After Reactor Trip.

(Includes Startup of An Inactive Reactor Coolant Loop).

The operations will be conducted without reliance on intermediate IC sets and using actual plant procedures. Every effort will be made to stay within normal operations guidelines and procedures however, minor deviations may be made to expedite the test such as exceeding the 3%/hr loading limit. This phase of testing will identify those discrepancies that may exist between the simulator and V.C. Summer Nuclear Station and verify that the simulator can be operated using actual plant procedures. These tests will be performed at least once every 4 years in the middle of core life conditions.

1.4.1 Acceptance Criteria for Normal Operations Testing

a. The response of the simulator shall be compared to V.C. Summer Nuclear Station plant response or best estimate plant response, and acceptance criteria, where applicable, shall be the same as plant startup/transient test procedure acceptance criteria.

b. The simulator will not fail to alarm or cause an automatic action if the V.C. Summer Nuclear Station would have caused an alarm or automatic action.

c. The simulator will not cause an alarm or automatic action if the V.C. Summer Nuclear Station would not cause an alarm or automatic action.

d. The observable change in simulator parameters will be in the same direction as "best estimate" analysis and will not violate the physical laws of nature.

e. The simulator can be operated in accordance with the V.C. Summer Nuclear Station General Operating Procedures.

f. Where responses or indications deviate from those specified in the test procedure the appropriate test data sheet(s) will be marked to reflect those deviations and the unexpected indication or deviation evaluated according to the normal operations test acceptance criteria by the Simulator Operations Specialists.

1. If the deviation is due to a procedure error the test shall be marked satisfactory and the test procedure revised.

2. If the deviation is due to a minor simulation problem (i.e., does not compromise the intent of the test), the test shall be marked satisfactory and an SDR written to document the unsatisfactory response.

3. If the deviation is due to a plant modification (MRF) or a simulator/simulation problem a Simulator Discrepancy Report (SDR) shall be written and the test marked unsatisfactory. The test will remain unsatisfactory until the SDR has been cleared and the test has been satisfactorily completed.

1.5 Malfunction Testing

The purpose of malfunction testing is to demonstrate the proper simulator response to the initiation of a specific malfunction. In some instances, best engineering judgement will be used based on operator/instructor experience as to whether a particular malfunction response is realistic and logical. As V.C. Summer Nuclear Station data becomes available through special plant tests, modification program, off-normal occurrences or significant plant operating events, the simulator is tested to verify that it properly mimics the plant data.

a. The following list details, but does not limit, those documents used in supporting the desired plant response:

1. Precautions, Limitations, and Setpoint Document.

2. Plant Operations Package, V.C. Summer Unit 1, Nuclear Power Cycle 1, WCAP-9854 Nov. 1980. Ammend to Cycle 3, WCAP-10874 March 1986.

3. Core Physics Characteristics of Westinghouse Three Loop Nuclear Plants (Unit 1, Cycle 1), WCAP-8102-5, June 1977.

4. Nuclear Design of V.C. Summer Unit 1, Cycle 1, WCAP-9685, March 1980.

5. Instrument Setpoint Document.

6. Design Transient Manual.

7. Startup Test Manual.

8. FSAR.

9. System Descriptions.

10. Technical Manuals.

11. GAI Drawings.

12. Plant Process Computer Logs, Print Value Reviews, Post Trip Reviews, Trend Logs and Alarm Printouts.

13. General Operating Procedures, Emergency Operating Procedures, System Operating Procedures, Surveillance Operating Procedures.

14. Off-Normal Occurrences and Significant Plant Operating Events.

15. Plant Modification Program (MRF).

b. Malfunction Testing Frequency

Some malfunctions will be tested during different core ages since the core response will be dependent upon core age. All certified malfunctions will be tested every 4 years. The simulator test plan will identify all annual testing requirements except special tests.

c. Multiple Concurrent Malfunctions

Malfunctions are tested on a one-at-a-time basis so that both the gross and subtle simulator responses can be evaluated in a controlled and known situation. Confidence in the individual malfunctions, coupled with the fact that the modeling of the central system is based on the dynamic equations of state, leads to confidence that the interactions of multiple concurrent malfunctions will be correct to within the specified tolerances. This is monitored during simulator training sessions and SDRs are initiated when discrepancies are identified when multiple concurrent malfunctions are used.

1.5.1 Acceptance Criteria for Malfunction Testing

a. General

5.1 The response of the simulator shall be compared to V.C. Summer Nuclear Station plant response or best estimate plant response, and acceptance criteria, where applicable, shall be the same as plant startup/transient test procedure acceptance criteria.

5.2 The simulator will not fail to alarm or cause an automatic action if the V.C. Summer Nuclear Station would have caused an alarm or automatic action.

5.3 The simulator will not cause an alarm or automatic action if the V.C. Summer Nuclear Station would not cause an alarm or automatic action.

5.4 The observable changes in simulator parameters shall be in the same direction as "best estimate" analysis and do not violate the physical laws of nature. "Best Estimate" shall be defined as:

a) V.C. Summer Nuclear Station data if available, or

b) Similar reference plant data if available, or

c) Evaluated as "best estimate" in the judgement of the Simulator Test Certification Panel. Evaluations as "best estimate" shall only be made after a review and careful consideration of plant status, reference material and plant modifications by the Simulator Operations Specialists.

5.5 The simulator's response will be consistent with the expected indications and responses as detailed on Attachment I.

b. Specific

1. The malfunction test procedure specifies and details the appropriate simulator response. It identifies:

   a) Which alarms, status lights, annunciators and control or protective functions must occur.

   b) Minimum operator actions necessary for the satisfactory completion of the test.

2. The simulator's response shall be consistent with the expected indications and responses as detailed in the malfunction test procedure.

3. Where responses or indications deviate from those specified in the test procedure, the appropriate expected response verification sheet(s) will be marked to reflect those deviations. The unexpected indication or deviation will be evaluated according to the malfunctions test general acceptance criteria by the Simulator Operations Specialists.

   a) If the deviation is due to a procedure error, the test shall be marked satisfactory and the test procedure revised.

   b) If the deviation is due to a minor simulation problem (ie.,does not compromise the intent of the test), the test shall be marked satisfactory and an SDR written to document the unsatisfactory response.

   c) If the deviation is due to a plant modification (MRF) or a simulator/simulation problem, an SDR shall be written and the test marked unsatisfactory. The test will remain unsatisfactory until the SDR has been cleared and the test has been satisfactorily completed.

   d) If the unexpected indication or deviation occurs on a certified malfunction test and is believed to be valid, the

test must be reviewed and recertified by the Simulator Test Certification Panel. Otherwise, an SDR must be written to document the unsatisfactory response. The test will remain unsatisfactory until the SDR has been cleared and the test has been satisfactorily completed.

1.6 Transient Testing

Transient testing is divided into four categories: Normal Transient Testing, Casualty Transient Testing, FSAR Transient Testing, and LER Testing. Transient Testing is designed to impose transients encountered during normal plant operations or abnormal transients encountered during accident conditions, including multiple failures, then verifying the simulators response matches or coincides with that expected from the V.C. Summer Nuclear Station.

a. Normal Operations Transient Testing

Types of normal operation transient testing are listed below. These tests will be run as a part of normal operations testing.

1. RCP Startup and Shutdown.
2. Plant Heatup and Cooldown.
3. Unit Loading and Unloading Between 0 and 100 Percent Full Power (includes 10%/step load change).
4. Plant Recovery From Rx Trip to 100 Percent Power (includes startup of an inactive reactor coolant loop).
5. Load Rejection (up to full).

b. Casualty Transient Testing (Appendix B Tests)

1. Simultaneous Closure of All Three MSIVs.
2. Simultaneous Trip of All Three RCPs.
3. Main Turbine Trip (at maximum power level which does not cause Rx Trip).
4. 100% LOCA with Loss of Offsite Power.
5. 100% Unisolable Main Steamline Break.
6. Slow Primary System Depressurization to Saturated Conditions (ECCS Inhibited).
7. Manual Reactor Trip
8. Simultaneous Trip of All Three FWPs
9. Trip of One RCP.

10. Maximum Rate Power Ramp to 75% Back to 100%.

c. Final Safety Analysis Report Testing (FSAR)

The simulator response will be compared to the accidents listed in Chapter 15 of the Final Safety Analysis Report. Safety analysis calculated response is based upon conservative initial conditions and assumptions, and may not accurately reflect realistic plant response. Therefore, it would not be practical to attempt to duplicate the initial conditions and assumptions, or to compare the simulator response to the FSAR results. Final Safety Analysis Report (FSAR) testing will be conducted as a part of Normal Operations Testing, Malfunction Testing, or Casualty Transient Testing. The tests will be conducted under realistic plant conditions and the results compared to realistic plant response, which should produce results less severe than FSAR results. The FSAR transients are:

1. Condition I: Normal operation and Operational Transients. These tests will be run as a part of normal operations testing.

    a) Power Operation (15 to 100 percent of full power).

    b) Startup (critical, 0 to 15 percent of full power).

    c) Hot Shutdown (subcritical, residual heat removal system isolated).

    d) Cold Shutdown (subcritical, residual heat removal system in operation).

    e) Refueling (mode 5, 140°F, depressurized).

2. Condition II: Faults of Moderate Frequency (Malfunction Tests).

    a) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition.

    b) Rod Cluster Control Assembly Misalignment.

    c) Uncontrolled Boron Dilution.

    d) Partial Loss of Forced Reactor Coolant Flow.

    e) Startup of an Inactive Loop.

    f) Loss of External Electrical load and/or Turbine Trip.

    g) Loss of Normal Feedwater.

    h) Loss of Offsite power to Station Auxiliaries.

    i) Excessive Heat Removal Due to Feedwater System Malfunction.

    j) Excessive Load Increase Incident.

    k) Accidental Depressurization of the RCS.

    l) Accidental Depressurization of the Main Steam System.

m) Inadvertent Operation of ECCS System During Power.

3. Condition Three Events: Infrequent Faults (Malfunction Tests).

a) Complete Loss of Forced Reactor Coolant Flow.

4 Condition IV Events: Limiting Faults (Malfunction Tests).

a) Major Rupture of a Steamline.

b) Major Rupture of a Main Feedwater Pipe.

c) Single Reactor Coolant Pump Locked Rotor.

d) Rupture of a Control Rod Drive Mechanism Housing.

d. <u>LER Testing</u>

Off-Normal Occurrences at the V.C. Summer Nuclear Station are documented by SAP-132, Off-Normal Occurrence Evaluation, Reporting and Resolution. This document, with attachments, provides sufficient information to allow comparison of V.C. Summer Nuclear Station data under abnormal conditions with the simulator. An informal test procedure will be developed which, to the extent possible, will duplicate the reactor and plant conditions existing before the event. The particular abnormal condition will be introduced via LOA's, malfunctions or operator actions in as nearly as identical fashion as that occurred at the V.C. Summer Nuclear Station, with the same operator actions and at the same times and conditions as can be determined.

1.6.1 Acceptance Criteria for Transient Testing

a) Acceptance Criteria for Normal Operations Transient Testing.

Acceptance criteria is the same as Section 1.4.1, Normal Operations Testing.

b) Acceptance Criteria for Casualty Transient Testing.

Acceptance criteria is the same as for Section 1.5.1, Acceptance Criteria for Malfunction Testing.

c) Acceptance Criteria for FSAR Testing

Final Safety Analysis Report (FSAR) Testing will be accomplished as a part of Normal Operations Testing, Malfunction Testing, or Casualty Transient Testing. Therefore, FSAR Acceptance Criteria will be governed according to where each FSAR test is placed in the Integrated System Test Plan. Simulator response will be compared to expected response based upon FSAR, Chapter 15.

d) Acceptance Criteria for LER Testing

1. Acceptance Criteria for LER Testing shall be that the response exhibited by the simulator corresponds to the data contained in the Off-Normal Occurrence package or,

2) Will be evaluated according to the malfunction test procedure general acceptance criteria.

e. Transient Testing Frequency

1) Normal Operations Transient testing is performed using a schedule that repeats every four years.

2) Casualty Transient testing is performed on an annual basis.

3) FSAR Malfunction testing is performed using a schedule that repeats every four years.

4) Plant LER testing is performed using a schedule that repeats every four years.

5) Real Time testing of the simulator is performed annually.

1.7 Surveillance Testing

Selected surveillance tests have been identified in the integrated system test plan. These tests will be conducted utilizing the appropriate surveillance test procedure. Acceptance criteria will be the same as that specified by the appropriate surveillance test procedure.

1.8 Main Control Board

The Main Control Board Audit verifies the simulator control panels duplicate the main control board panels of the V.C. Summer Nuclear Station. The audit is conducted by comparing the simulator to the V.C. Summer Station Main Control Board Panel color drawings, color photographs and video tapes of the main plant control room.

2.0 TEST PREPARATION AND CONDUCT

Testing is normally conducted by, or under the direction of, the Simulator Operations Specialists but may be conducted by other members of the Nuclear Operations Education and Training Staff.

2.1 Test Personnel Requirements

Personnel performing tests on the V.C. Summer Simulator must be:

a. Simulator Operations Specialists, or

b. Licensed/Certified Senior Reactor Operator Instructors, or

c. Other individuals who have qualified as a simulator instructor as specified in NTCI-401, Simulator Training and Evaluation, and who hold, or have held, a Senior Reactor Operator License.

2.2 Conduct of Operations

Operations testing will be conducted in an organized and professional manner at all times. Conduct should reflect that conduct normally expected and required in the V.C. Summer Nuclear Plant Control Room. If a test team is assembled to conduct a test, at least one member must meet the requirements of Section 2.1 and will be designated the Test Coordinator. The Test Coordinator or test performer as specified in Section 2.1 is responsible for:

a. Reviewing and understanding all tests to be conducted.

b. Obtaining required copies of blank attachments, data sheets or the supporting documents prior to the start of each test.

c. Conducting the test.

d. Ensuring all data is collected and signoffs are complete.

e. Ensuring SDRs are written when deviations or problems are identified during the test.

f. Forwarding test procedure(s) with all attachments, data sheets, discrepancy reports and printouts to the Simulator Operations Specialists.

The other members of the test team may function as plant operators or data takers. Plant operators should perform their duties in accordance with general, emergency and system operating procedures, as modified by the test procedure, to dictate the level of operator response desired in order to successfully conduct and complete the required tasks.

2.3 Test Methods

The following recommended test methods should be followed in order to obtain accurate information:

a. At the start of the test, set the cumulative computer run time (MTIME) to zero.

b. If an unanticipated abnormal operation or computer abort occurs, press FRZ and record the cumulative computer run time (upper left of the Instructor Console CRT) for later reference.

c. When the cause of the trip or abnormality has been corrected, simulation will continue with MTIME set to the age of the BACKUP SNAP used to re-initialize simulation. The test will resume when MTIME equals the MTIME recorded in Step b, above.

d. If Plant Process Computer Post Trip Reviews are required for testing purposes set JIPCPLOG R = T after each initialization.

e. The simulator chart recorders are not used during most simulator testing. If needed, turn them on/off from the instructors booth.

f. All recording of data will be with BLACK INK BALLPOINT PEN. If an error is made during recording of data, place a single line through the error, initial the strikeout, and record the correct data adjacent to the original recording.

3.0 TESTING

Simulator testing falls into two(2) categories, annual testing governed by the Simulator Test Plan and special tests that arise as a result of Off-Normal Occurrences, LERs and Industry Operating Experience Reports.

3.1 Simulator Test Plan

A simulator annual operability test is required by the ANSI/ANS-3.5, comprised of steady state performance tests and transient tests of a benchmark set of transients. In addition, the simulator real time test will be performed annually.

Regulatory Guide 1.149, Nuclear Power Plant Simulation Facilities For Use in Operator License Examinations, requires that all the certified simulator malfunctions be tested in their entirety over a four year period. This is in addition to the required annual operability test. Exceptions to the 4-year plan may be necessary due to:

a. Core modeling changes.

b. Plant modifications.

Special tests such as plant tests, and Licensee Event Reports (LERs) and Off-Normal Occurrences (ONOs) will be performed as soon as possible after the event occurred.

3.2 Simulator Test Abstracts

3.2.1 Simulator test abstracts are developed and completed for each of the certified tests specified in ANSI/ANS-3.5, including Appendices.

3.2.2 Test abstracts contain the following:

- Date test conducted.
- Name and description of the test.
- Available options.
- Tested options.
- Initial conditions.
- Final conditions.
- Description of baseline data used to determine fidelity to V. C. Summer.
- Deficiencies found, corrective action planned and dated by which corrections will be made.
- Exceptions taken to ANSI/ANS-3.5 as a result of the test, with justification.

NOTE:

Simulator Test Certification Panel specifies the correct response for malfunctions where no reference plant data exists.

3.2.3 Test Abstracts will be filed with the completed test results until the next simulator certification report is generated. They will then be attached to the certification report.

4.0 SIMULATOR DISCREPANCY REPORT (SDR)

4.1 Whenever a hardware or software problem is discovered during testing, a Simulator Discrepancy Report (SDR) must be written. The SDR should be as specific as possible in describing the problem and what the expected results from the simulator should be. The SDR number must be recorded on the applicable Test Data Sheet or Expected Response Verification Form.

5.0 TEST DOCUMENTATION

5.1 At the conclusion of each test, the test data sheet or the expected response verification sheet shall be filled out and signed by the Test Performer or Test Coordinator. The Test Performer or Test Coordinator should ensure that the test is complete and that all test documents have been collected. Test documents may include:

a. Test Data Sheets.

b. Expected Response Verification Sheet.

c. Line Printer Printouts (where required).

d. Line Recorder Printouts (where required).

e. Various Plant Computer Printouts (where required).

f. Simulator Discrepancy Reports (SDRs).

g. Simulator Test Abstract.

5.2 Document Retention

Test documents will be retained for a minimum of 2 complete test cycles (8 years).

6.0 RETESTING

6.1 Satisfactory completion of the test is contingent upon the following:

a. All test objectives and acceptance criteria are met.

b. All required test documentation is complete and attached.

c. Either there are no SDRs, or only minor SDRs, that would not compromise the intent of the test, or

d. All SDRs not characterized as minor have been satisfactorily retested and the test has been satisfactorily rerun.

6.2 Simulator tests involving only minor SDR's need not be repeated. Normal SDR testing to resolve minor simulator problems is sufficient.

7.0 REFERENCES

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

2. Nuclear Regulatory Guide 1.149-1985, Nuclear Power Plant Simulators for Use in Operator Training.

3. 10CFR55.45(b), Operator Licenses 5/26/87.

4. Nuclear Regulatory Guide 1258, Evaluation Procedure for Simulation Facilities Certified Under 10CFR55.

5. Westinghouse Factory Acceptance Program for V.C. Summer Nuclear Station.

6. Nuclear Training Center Instruction Two (NTCI-2), Review of MRFs, MCNs, and Special Instructions.

7. NTCI-5, Simulator Discrepancy Reports.

8. NTIC-401, Simulator Training and Evaluation.

9. V.C. Summer Nuclear Station Final Safety Analysis Report (FSAR), Chapter 15.

10. Testrx Program For V.C. Summer Simulator, Revision 1, 9/26/86.

3.0.2 Discussion of Exceptions to ANSI/ANS 3.5-1985 Required Testing (References made herein are to Sections of the ANSI/ANS Standard)

Section 3.1.1 - Normal Plant Evolutions

Item (9): "Core performance testing . . . using permanently installed instrumentation."

Exception: The Virgil C. Summer Nuclear Station does not have "permanently installed" instrumentation to perform these tests. For that reason these tests are not supported on the simulator. We have performed the required testing using the TESTRX Program, supplied by Westinghouse. The plant gathers the same data using a temporarily installed reactivity computer. Both sets of data are on file and have been satisfactorily compared in an on-site acceptance test for the TESTRX Program.

Section 3.1.2 - Plant Malfunctions

Comparison to the Safety Analyses of FSAR Section 15.2

- Uncontrolled RCCA bank withdrawal at power.

Exception: Due to the fact that plant Axial Flux Difference operating limits constrain rod position at high powers to the nearly full out condition, completely withdrawing the rods would have little effect. Hence, a more challenging event is uncontrolled rod withdrawal from the just critical condition. IST-6.4.6.2 is designed to test this event.

- Uncontrolled Dilution During Refueling
- Uncontrolled Dilution During Cold Shutdown
- Uncontrolled Dilution During Startup
- Uncontrolled Dilution at Full Power

Exception: Of the four dilution events, a dilution event at Hot Standby presents the shortest time from alerting the operator to a loss of shutdown margin and of potential reactor criticality. This event, therefore, is selected for certification testing. All dilution events (except for dilution with the Reactor Vessel head removed) are capable of being simulated.

Comparison to FSAR Section 15.3 Analyses

- Minor Secondary System Pipe Breaks

Exception: Secondary pipe breaks of any size are capable of being simulated on the simulator. The FSAR provides an analysis for the Design Basis Main Steam line break and for an Accidental Depressurization of the Main Steam System (SG Safety Valve Failure). IST-7.5 and IST-6.8.10, respectively, duplicate the FSAR tests. A specific FSAR analysis of minor secondary system pipe breaks is not done; therefore, a specific certification test is not performed.

- Inadvertent Loading of a Fuel Assembly into an Improper Position

Exception: Operations during a refueling condition is not a licensed operator simulator training evolution and is, therefore, not modeled or tested.

Nuclear Operations Education & Training conducts limited training sessions for Refueling Senior Reactor Operators, but, training focuses primarily on responsibilities and refueling procedures, and it is conducted entirely in the classroom.

- Waste Gas Decay Tank Rupture

Exception: Since this event is not a licensed operator simulator training evolution, it is not modeled or tested. Because of limited indications available on the control board for a rupture of a waste gas decay tank, using the simulator for this training exercise would be inefficient utilization of training time. Hence, the training focus is primarily on classroom instruction.

Comparison to FSAR Section 15.3 Analyses

- Single RCCA Withdrawal at Full Power

Exception: Since no single electrical or mechanical failure could cause this event, it is not modeled or tested.

Comparison to FSAR Section 15.4 Analyses

- Fuel Handling Accid nt

Exception: Since this event is not a licensed operator simulator training evolution, it is not modeled or tested. There are no specific tasks for initial or requalification training for licensed operators which address fuel handling accidents.

Section 3.1.2 - Plant Malfunctions

Item (12): "Control rod failure(s)." This item refers to "drifting " and "uncoupled" rods. Stuck, misaligned, and dropped rods are capable of being simulated.

Exception: Due to the Westinghouse PWR configuration, these are not considered as credible events and are, therefore, not modeled or tested on the simulator. The only time rods are uncoupled is during removal of the upper core internals during station refueling operations. Refueling is not simulated on the Virgil C. Summer Simulator.

There are no tasks for initial or requalification training that require addressing "drifting or uncoupled" rods. Nuclear Operations Education & Training conducts limited training sessions for Refueling Senior Reactor Operators; but training focuses primarily on responsibilities and refueling procedures, and it is conducted entirely in the classroom.

3.1 Simulator Real Time Test Abstract

This section contains the test abstract of the simulator real time test. Real time simulation capability is required per Sections 3.1.1 and 3.1.2 of ANSI/ANS 3.5-1985.

The simulator real time test was completed satisfactorily during initial certification testing.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

IST-11.1, *Simulator Real Time Test*

Description of Test:

This test demonstrates the capability of the simulator dynamic models to run in real time per Appendix A of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator performance of time-controlled events is compared to the real time requirements of the Virgil C. Summer Nuclear Station Surveillance Test Procedure. Brush recorder traces are taken to record the real time completion of the monitored events.

Malfunction Description:

Malfunctions EPS-5A, "Loss of ESF Bus 1DA Normal Feed Breaker", and RCS-5, "Large LOCA in Cold Leg", are used for this test. Malfunction EPS-5A causes a blackout on Bus 1DA and an automatic start of Emergency Diesel Generator 1A. The LOCA, Malfunction RCS-5, causes a Safety Injection actuation.

The Blackout and Safety Injection will cause the automatic starting of ESF equipment, controlled by the ESF Load Sequencer.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the event is started. The simulation is stopped after all ESF equipment has been started and the starting times recorded.

Date of Test:

March 1, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Surveillance Test Procedure STP-125.010 *Integrated Safeguards Test Train A.*

Test Results:

The test was completed satisfactorily with some minor procedural problems which did not affect the results.

Corrective Actions: (if necessary)

None.

Comments:

None.

3.2 Steady State and Normal Operations Tests Abstracts

This section contains the test abstracts of the steady state and normal operations tests. The capability to simulate steady state and normal operations, including operator conducted surveillance testing [illegible] safety-related equipment or systems, is required per Section 3.1.1 of ANSI/ANS 3.5-1985.

The steady state and normal operations tests were completed satisfactorily during initial certification testing.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.1, *Power Reduction To 75% Power.*

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. The simulator is taken from a 100% power condition to a 75% power condition using the Virgil C. Summer Nuclear Station Operating Procedures. In addition, 10% step load changes are performed to verify the simulator's control systems response to design basis load changes.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized at the reference 100% power condition and the step swing tests are performed (100% to 90% to 100%). The simulator is then ramped down to a final power level of 75% and this condition is saved.

Date of Test:

November 19, 1990

Baseline Data References:

General Operating Procedure GOP-4, *Power Operation.*
Power Operation Test POT-7.3, *Load Swing Test At 100%*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.2, *Power Reduction To 50% Power*.

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. The simulator is taken from 75% power to 50% power using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized at the 75% power condition saved from IST-2.1. The simulator is then ramped down to 50% power and this condition is saved.

Date of Test:

November 19, 1990

Baseline Data References:

General Operating Procedure GOP-4, *Power Operation*.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.3, *50% Power To 2% Power*.

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. The simulator is taken from 50% power to 2% power using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized at the 50% power condition saved from IST-2.2. The simulator is then ramped down to 2% power and this condition is saved.

Date of Test:

November 20, 1990

Baseline Data References:

General Operating Procedure GOP-4, *Power Operation*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.4, *2% Power To Hot Standby*.

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. The simulator is taken from 2% power to Hot Standby conditions using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized at the 2% power condition saved from IST-2.3 The simulator then has the reactor shutdown and Hot Standby conditions established. The condition is saved for future Normal Operations testing.

Date of Test:

November 20, 1990

Baseline Data References:

General Operating Procedure GOP-5, *Reactor Shutdown From Startup To Hot Standby (Mode 2 To Mode 3)*.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.5, *Hot Standby To Hot Shutdown*.

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. The simulator is cooled down from the Hot Standby condition to the Hot Shutdown condition using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the Hot Standby condition saved from IST-2.4. A cooldown is then initiated to the Hot Shutdown condition, and this condition is saved.

Date of Test:

November 21, 1990

Baseline Data References:

General Operating Procedure GOP-6, *Plant Shutdown From Hot Standby To Hot Shutdown (Mode 3 To Mode 4)*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.6, *Shutdown And Cooldown To Cold Shutdown*

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. The simulator is cooled down to the Cold Shutdown condition using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the Hot Shutdown condition saved from IST-2.5. A cooldown and depressurization to the Cold Shutdown condition is accomplished, and this condition is saved.

Date of Test:

December 1, 1990

Baseline Data References:

General Operating Procedure GOP-7, *Plant Shutdown And Cooldown From Hot Shutdown To Cold Shutdown (Mode 4 To Mode 5).*

Test Results:

The test was completed satisfactorily, but a minor problem of slow RCS dilution was observed.

Corrective Actions: (if necessary)

Trouble Report #365 was written on the dilution problem. The problem was corrected and tested satisfactorily on December 8, 1990.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.7, *Cold Shutdown To Hot Shutdown*

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. A heatup to the Hot Shutdown condition is simulated using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the Cold Shutdown condition saved from IST-2.6. A plant heatup is then conducted up to the Hot Shutdown condition and this condition is saved.

Date of Test:

December 19, 1990

Baseline Data References:

General Operating Procedure GOP-1, *Plant Startup And Heatup From Cold Shutdown To Hot Shutdown (Mode 5 To Mode 4)*

Test Results:

The test was completed satisfactorily with two minor problems observed: The pressurizer steam space temperature indication was discovered to be incorrect, and a steam flow transient was observed when first opening the main steam isolation valve bypass valve.

Corrective Actions: (if necessary)

1. Trouble Report #357 was written to correct pressurizer steam space temperature. The problem was corrected and tested satisfactorily on January 3, 1991.

2. Trouble Report #394 was written on the steam flow transient. The problem was corrected and tested satisfactorily on January 5, 1991.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.8, *Hot Shutdown To Hot Standby*.

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. A plant heatup from Hot Shutdown to Hot Standby conditions is performed using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the Hot Shutdown condition saved from IST-2.7. A heatup to Hot Standby conditions is conducted and this condition is saved.

Date of Test:

December 20, 1991

Baseline Data References:

General Operating Procedure GOP-2, *Plant Heatup From Hot Shutdown To Hot Standby (Mode 4 To Mode 3)*.

Test Results:

The test was completed satisfactorily with two minor problems noted. The RHR system cooldown after isolation was observed to be wrong and a problem with the functioning of the Steam Generator Power Operated Relief Valve controllers was noted.

Corrective Actions: (if necessary)

1. Trouble Report #397 was written on the controllers. The problem was corrected and tested satisfactorily on January 5, 1991.
2. Trouble Report #398 was written to correct the RHR cooldown. The problem was corrected and tested satisfactorily on February 13, 1991.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.9, *Hot Standby To 2% Power*.

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. A reactor startup is performed and power is raised to 2% using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the Hot Standby condition saved from IST-2.8. A reactor startup is conducted and power is raised to the 2% power condition. This condition is saved for future Normal Operations testing.

Date of Test:

January 11, 1991

Baseline Data References:

General Operating Procedure GOP-3, *Reactor Startup From Hot Standby To Startup (Mode 3 To Mode 2)*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.10, *2% Power To 100% Power.*

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. Power escalation from 2% power to 100% power is performed using the Virgil C. Summer Nuclear Station Operating Procedures.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the 2% power condition saved from IST-2.9. A plant startup and power escalation to 100% power is conducted. This condition is saved for subsequent Normal Operations testing.

Date of Test:

December 21, 1990.

Baseline Data References:

General Operating Procedure GOP-4, *Power Operation (Mode 1).*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Normal Operations Test IST-2.11, *Recovery To Rated Power After Reactor Trip*

Description of Test:

This test demonstrates the ability to conduct normal plant evolutions on the simulator per Section 3.1.1 of ANSI/ANS 3.5-1985. A recovery to rated power after reactor trip is performed using the Virgil C. Summer Nuclear Station Operating Procedures. In addition, operation at less than full reactor coolant flow is also demonstrated. The safety analysis test of "Startup of an Inactive Reactor Coolant Pump" is also performed in this test procedure.

Malfunction Description:

Not applicable

Test Conditions:

The simulator is initialized in the 100% power condition and the reactor is tripped. The reactor is restarted and power raised to 35% where the B RCP is tripped to demonstrate operation at less than rated flow. The "Startup of an Inactive Reactor Coolant Pump" test is then performed. The simulator is re-initialized to the 35% power condition with all RCPs running and power is raised to 100%, the final condition.

Date of Test

December 26, 1990

Baseline Data References

1. Virgil C. Summer Nuclear Station General Operating Procedures.
2. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
3. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.

Test Results:

The test was completed satisfactorily with the exception of a poor quality brush recorder trace on the "startup of inactive loop" portion of the test.

Corrective Actions: (if necessary)

The "startup of inactive loop" portion of the test was performed again on December 28, 1990 and a satisfactory brush recorder trace obtained

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Steady State Performance Test IST-4.1, *100 Percent Power Heat Balance and Stability Test.*

Description of Test:

This test performs the 100% power heat balance test per Appendix B of ANSI/ANS 3.5-1985. A comparison of critical and non-critical parameters to the plant values is made, and a set of Technical Specification log readings is taken. The heat balance is calculated using the Virgil C. Summer Nuclear Station Surveillance Test Procedure. Mass balances are calculated for the pressurizer, the steam generators, and the condensate to deaerator. The simulator stability at 100% power is also verified.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at 100% power.

Date of Test:

January 2[illegible], [illegible]1

Baseline Data References:

1. Virgil C. Summer Nuclear Station Plant Computer Print Value Review Printout.
2. STP-102.002, *NIS Power Range Heat Balances Surveillance Test Procedure.*
3. Special Instruction SI-87-03, *Operating Logs.*

Test Results:

The test was completed satisfactorily with some minor procedural problems which did not affect the results.

Corrective Actions: (if necessary)

A procedure change was initiated to resolve these problems.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Steady State Performance Test IST-4.2, *75 Percent Power Heat Balance and Stability Test.*

Description of Test:

This test performs the 75% power heat balance test per Appendix B of ANSI/ANS 3.5-1985. A comparison of critical and non-critical parameters to the plant values is made, and a set of Technical Specification log readings is taken. The heat balance is calculated using the Virgil C. Summer Nuclear Station Surveillance Test Procedure. Mass balances are calculated for the pressurizer, the steam generators, and the condensate deaerator. The simulator stability at 75% power is also verified.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at 75% power.

Date of Test:

December 27, 1990

Baseline Data References:

1. Virgil C. Summer Nuclear Station Plant Computer Print Value Review Printout.
2. STP-102.002, *NIS Power Range Heat Balances Surveillance Test Procedure.*
3. Special Instruction SI-87-03, *Operating Logs.*

Test Results:

The test was completed satisfactorily with some minor procedural problems which did not affect the results.

Corrective Actions: (if necessary)

A procedure change was initiated to resolve these problems.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Steady State Performance Test IST-4.3, *50 Percent Power Heat Balance and Stability Test.*

Description of Test:

This test performs the 50% power heat balance test per Appendix B of ANSI/ANS 3.5-1985. A comparison of critical and non-critical parameters to the plant values is made, and a set of Technical Specification log readings is taken. The heat balance is calculated using the Virgil C. Summer Nuclear Station Surveillance Test Procedure. Mass balances are calculated for the pressurizer, the steam generators, and the condensate deaerator. The simulator stability at 50% power is also verified.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at 50% power.

Date of Test:

December 27, 1990

Baseline Data References:

1. Virgil C. Summer Nuclear Station Plant Computer Print Value Review Printout.
2. STP-102.002, *NIS Power Range Heat Balances Surveillance Test Procedure*
3. Special Instruction SI-87-03, *Operating Logs.*

Test Results:

The test was completed satisfactorily with some minor procedural problems which did not affect the results.

Corrective Actions: (if necessary)

A procedure change was initiated to resolve these problems.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.1.1, *Source Range Operability Test STP-102.001*.

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of theSource Range Nuclear Instrumentation channels using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at Hot Standby conditions.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-102.001, *Source Range Operability Test*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.1.2, *Nuclear Instrumentation System (NIS) Power Range Heat Balance.*

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 ANSI/ANS 3.5-1985. This test performs a calibration of the Nuclear Instrumentation System (NIS) power range channels by calculating a heat balance. The test is performed using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

December 21, 1990

Baseline Data References:

1. STP-102.002, *NIS Power Range Heat Balance.*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions(if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.1.3, *Intermediate Range Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 ANSI/ANS 3.5-1985. This test verifies the operability of the Intermediate Range Nuclear Instrumentation channels using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not . :able.

Test itions:

The tes. . conducted at Hot Standby conditions.

Date of Test:

January 8, 1991

Baseline Data References:

1. STP-102.003, *Intermediate Range Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions(if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.2.1, *CVCS Valve Operability Test*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of CVCS valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 5, 1991

Baseline Data References:

1. STP-105.002, *CVCS Valve Operability Test*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.2.2, *Safety Injection Valve Operability Test*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of safety injection valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at the reference 100% power condition.

Date of Test:

January 7, 1991

Baseline Data References:

1. STP-105.003, *Safety Injection Valve Operability Test*.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.2.4, *RHR System Valve Operability Test*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of the RHR System valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 5, 1991

Baseline Data References:

1. STP-105.005, *RHR System Valve Operability Test*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.3.1, *Movable Rod Insertion Test*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test establishes the operability of the control rods using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-106.001, *Rod Insertion Test*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.4.1, *Quadrant Power Tilt Ratio*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test calculates the quadrant power tilt ratio using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-108.001, *Quadrant Power Tilt Ratio*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.5.1, *RB Spray System Valve Operability Test*.

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of RB Spray System valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at the reference 100% power condition.

Date of Test:

January 5, 1991

Baseline Data References:

1. STP-102.003, *RB Spray System Valve Operability Test*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.6.1, *Operational Leakage to RCP Seals*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies that the operational leakage to RCP seals is within limits using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at the reference 100% power condition.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-114.001, *Operational Leakage to RCP Seals*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.6.2, *Operational Leak Test*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies that the operational leakage from the RCS is within limits using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at the reference 100% power condition.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-114.002, *Operational Leak Test*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.7.1, *Reactor Building Cooling Unit Functional Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of the reactor building cooling units using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-116.001, *Reactor Building Cooling Unit Functional Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.8.1, *Iodine Removal System Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of the Iodine Removal System using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-117.001, *Iodine Removal System Test.*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.9.1, *Main Steam Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of the main steam valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

January 5, 1991

Baseline Data References:

1. STP-121.00[illegible], *Main Steam Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.10.1, *Train A Service Water System Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of Train A se... ce water valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is performed at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-123.003A, *Train A Service Water System Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.10.2, *Train B Service Water System Valve Oper ability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of Train B service water valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-123.003B, *Train B Service Water System Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.11.1, *Control Room Emergency Air Cleanup System Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of the Control Room Emergency Air Cleanup system using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at at the reference 100% power condition.

Date of Test:

January 8, 1991

Baseline Data References:

1. STP-124.001, *Control Room Emergency Air Cleanup System Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.12.1, *Diesel Generator Operability Test*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of the diesel generator using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-125.002, *Diesel Generator Operability Test*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.13.1, *Pressurizer Block Valve Operability Test*

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of the pressurizer power operated relief valve block valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at at the reference 100% power condition.

Date of Test:

January 7, 1991

Baseline Data References:

1. STP-127.001, *Pressurizer Block Valve Operability Test*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.14.1, *Axial Flux Difference Calculation*.

Description of Test:

This test performs operator conducted surveillance testing on safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test calcualtes the axial flux difference using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

This test is conducted at the reference 100% power condition.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-133.001, *Axial Flux Difference Calculation*.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.15.1, *Shutdown Margin Verification*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. This test verifies the shutdown margin is greater than the Technical Specification requirement using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-134.001, *Shutdown Margin Verification*.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.16.1, *Steam Generator Blowdown Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. This test verifies the operability of the steam generator blowdown valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power conditions.

Date of Test:

January 7, 1991

Baseline Data References:

1. STP-136.001, *Steam Generator Blowdown Valve Operability Test.*

Test Results:

The initial test was unsatisfactory uncovering a problem with the opening times of PVG-503A, 503B, and 503C.

Corrective Actions (if necessary):

Simulator Discrepancy Report 91002 was written to correct the opening stroke times of PVG-503A, 503B, and 503C. The test was completed satisfactorily on January 17, 1991.

Comments:
None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.17.1, *Radiation Monitor Monthly Source Check*.

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of the atmospheric and liquid radiation monitors using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is performed at the reference 100% power condition.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-137.002, *Radiation Monitor Monthly Source Check*.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.18.1, *Post Accident Hydrogen Removal Valve Operability Test*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of post accident hydrogen removal valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable,.

Test Conditions:

The test is performed at the reference 100% power condition.

Date of Test:

January 7, 1991

Baseline Data References:

1. STP-138.001, *Post Accident Hydrogen Removal Valve Operability Test*

Test Results:

A problem with valve stroke times was discovered during the initial test.

Corrective Action (if necessary):

Simulator Discrepancy Report 91001 was written to correct valve stroke times. After correction, the test was completed satisfactorily on January 17, 1991.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.19.1, *RB And AB Nuclear Drains Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per Section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of the nuclear drain valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is performed at the reference 100% power condition.

Date of Test:

January 8, 1991

Baseline Data References:

1. STP-140.001, *RB And AB Nuclear Drains Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.20.1, *Reactor Coolant System Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of Reactor Coolant System valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is performed at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-142.001, *Reactor Coolant System Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.21.1, *Turbine Trip Actuating Device Operational Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operation of the turbine trip actuating devices using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at Hot Standby condition.

Date of Test:

January 10, 1991

Baseline Data References:

1. STP-142.005, *Turbine Trip Actuating Device Operational Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.22.1, *Nuclear Sampling Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of nuclear sampling valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-144.001, *Nuclear Sampling Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRG' . C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.23.1, *Waste Processing Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of waste processing valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-145.001, *Waste Processing Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.24.1, *Reactor Makeup Water Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of reactor makeup water valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is conducted at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-146.003, *Reactor Makeup Water Valve Operability Test*

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-8.25.1, *Feedwater Valve Operability Test.*

Description of Test:

This test performs operator conducted surveillance testing of safety related equipment per section 3.1 of ANSI/ANS 3.5-1985. The test verifies the operability of feedwater valves using the Virgil C. Summer Nuclear Station Surveillance Test Procedure.

Malfunction Description:

Not applicable.

Test Conditions:

The test is performed at the reference 100% power condition.

Date of Test:

January 9, 1991

Baseline Data References:

1. STP-148.001, *Feedwater Valve Operability Test.*

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

3.3 Transient Test Abstracts

This section contains the test abstracts of the transient tests. The simulator's capability to simulate plant transients is verified by performing a benchmark set of transients, specified in Appendix B of ANSI/ANS 3.5-1985, and by performing transients that have occurred at the referenced plant.

The transient tests were completed satisfactorily during initial certification testing.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.1, *Simultaneous Closure Of All 3 MSIVs.*

Description of Test:

The test demonstrates the capability of the simulator to simulate a simultaneous closure of all MSIVs per Appendix B of ANSI/ANS 3.5-1985. This test also performs the safety analysis transient of a loss of external load. The event is initiated at full power conditions and the simulator response is compared to the expected response. Brush recorder traces of the simulator response are taken to compare to the accident analysis curves.

The brush recorder traces are evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction PRS-14, "Inadvertent Main Steam Isolation", is used for this test. This malfunction simulates a failure that generates an inadvertent Train B Main Steam Isolation signal. All the MSIVs close simultaneously.

The plant response will be a rapid decrease in steam flow and main steam header pressure. Generator megawatts will also drop rapidly. RCS temperature and pressure will increase rapidly. Steam generator (SG) pressure will increase rapidly and the SG power-operated relief valves and safety valves will lift. Feed flow to the steam generators will drop as the feed pumps lose motive steam. The reactor will trip on low SG level or due to overtemperature, differential temperature.

Test Conditions:

The simulator is initialized in the 100% power beginning of life condition and the malfunction activated. The final conditions are in Hot Standby with the reactor tripped and decay heat removal through the SG power-operated relief valves.

Date of Test:

January 19, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.

2. Simulator Test Certification Panel.

Test Results:

The test was performed initially with some differences in response from that specified in the procedure.

Corrective Actions (if necessary):

The simulator response was evaluated and determined to be correct. A procedure change was submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.2, *Simultaneous Trip Of All 3 RCPs*

Description of Test:

This test demonstrates the capability of the simulator to simulate a simultaneous trip of all reactor coolant pumps per Appendix B of ANSI/ANS 3.5-1985. The test also performs the safety analysis transient of a complete loss of forced reactor coolant flow. The event is initiated at full power conditions and the simulator response compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves.

The brush recorder traces are evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction EPS-4, "Loss of Service Bus" is used for this test. This malfunction simulates the loss of the 7.2 KV busses which supply the RCPs. The RCPs are tripped by bus underfrequency and the reactor is tripped by bus underfrequency or undervoltage.

The plant response will be a reactor trip and cooldown to Hot Standby conditions.

Test Conditions:

The simulator is initialized in the 100% power beginning of life condition and the malfunctions activated. The final conditions are in Hot Standby on natural circulation.

Date of Test:

January 19, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15

2. Simulator Test Certification Panel

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.3, *Main Turbine Trip (At Maximum Power Level Which Does Not Cause Reactor Trip).*

Description of Test:

The test demonstrates the capability of the simulator to simulate a main turbine trip per Appendix B of ANSI/ANS 3.5-1985. The event is initiated at the maximum power level which will not result in a reactor trip. The simulator response is compared to the expected response. Manual actions are taken to stabilize conditions at approximately 5% power.

Malfunction Description:

Malfunction TUR-1, "Inadvertent Turbine Trip" is used for this test. This malfunction simulates a failure of the emergency trip valve which causes a turbine trip.

The plant response to this event will be an increase in RCS temperature and pressure and a decrease in steam flow. The control rods will rapidly step in and the steam dump valves will open to reduce RCS temperature. Reactor power will decrease to less than 10%. Manual actions are taken to stabilize power at approximately 5%.

Test Conditions:

The simulator is initialized in the 50% power condition (below P-9) and the malfunction activated. The final condition is at approximately 5% power with control rods in manual.

Date of Test:

January 20, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.4, *100% LOCA With Loss Of Offsite Power*

Description of Test:

The test demonstrates the capability of the simulator to simulate a 100% LOCA with a loss of offsite power per Appendix B of ANS-3.5. This test also performs the safety analysis transient of a design basis LOCA. The event is initiated at full power conditions, and the simulator response is compared to the expected response. Brush recorder traces are taken to compare with the accident analysis curves.

The brush recorder traces are evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

A number of different malfunctions are used for this test. Malfunction RCS-5, "Large LOCA In Cold Leg", provides a DBA size break in a selectable RCS loop. Malfunction EPS-3, "Loss of Emergency Auxiliary Transformer", causes the loss of the emergency auxiliary transformer. Malfunction EPS-5, "Loss of ESF Buss", causes a loss of power to the ESF buses. Malfunction EPS-9, "Loss of Unit Auxiliary Transformer", causes a loss of the unit auxiliary transformer. Actuation of these malfunctions simultaneously causes the LOCA with a loss of all offsite power.

The plant response will be a rapid RCS depressurization and partial core uncovery. Safety injection is actuated and the emergency diesel generators start and pick up the ESF loads. Reactor building temperature and pressure increase rapidly as well as radiation levels and sump water levels.

Test Conditions:

The simulator is initialized in the 100% power End of Life condition and the malfunctions activated. Manual actions are taken to align the ESF systems in cold leg recirculation mode, as directed by the Emergency Operating Procedures. The final conditions are in cold leg recirculation lineup with core cooling established.

Date of Test:

January 23, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
3. Simulator Test Certification Panel.

Test Results:

The test was performed successfully with only one minor difference from the procedure.

Corrective Actions (if necessary):

The difference was evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.5, *100% Unisolable Main Steamline Break*

Description of Test:

The test demonstrates the capability of the simulator to simulate a design basis unisolable main steamline break per Appendix B of ANSI/ANS 3.5-1985. The test also performs the safety analysis transient of the DBA steam line break. The event is initiated at Hot Standby conditions, the worst case scenerio for this accident; and the simulator response is compared to the plant response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves.

The brush recorder traces are evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction MSS-3,"Steamline Break Inside Containment", is used for this test. The malfunction allows the selection of any steamline inside containment to experience a break of up to 12E6 lb/hr ramped in at from 0 to 3600 seconds.

The plant response to a steamline rupture will be an increase in steam flow and rapidly decreasing RCS temperature and pressure. Reactor building pressure and temperature will increase and safety injection will be actuated. The main steam isolation valves will close, stopping steam flow from the unaffected steam generators. The affected steam generator will then rapidly blow down.

Test Conditions:

The simulator is initialized at the Hot Standby end of life conditions and a break of 4E6 lb/hr is initiated on B steamline. Manual actions are taken to isolate the affected steam generator, per the Emergency Operating Procedures. The final conditions are with safety injection actuated and the affected steam generator isolated and empty.

Date of Test:

January 19, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.

2. Virgil C. Summer Nuclear Station Emergency Operating Procedures.

3. Simulator Test Certification Panel.

Test Results:

The test was performed successfully with some differences from the procedure observed.

Corrective Actions (if necessary):

The differences were evaluated as a correct response and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.6, *Slow Primary System Depressurization to Saturated Conditions*

Description of Test:

The test demonstrates the capability of the simulator to simulate a slow primary system depressurization to saturated conditions with the ECCS inhibited per Appendix B of ANSI/ANS 3.5-1985. The event is initiated at full power conditions, and the simulator response is compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves. The brush recorder traces are evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunctions PRS-7, "Safety Valve Failure", and PCS-5, "Safety Injection Failure", are used for this test. Malfunction PRS-7 allows the failure of the B pressurizer safety valve from 0% to 100% open. Malfunction PCS-5 allows the selection of either or both SI trains to fail to actuate or inadvertently actuate. For this test, the SI signals fail to actuate.

The plant response will be a decrease in RCS pressure, a reactor trip, and a failure of SI to actuate. The core subcooling margin will decrease and reactor vessel wide range level will decrease. The RCS pressure will stabilize at saturation pressure for the RCS hot leg temperature. The reactor coolant pumps will start cavitating as saturation conditions in the RCS are approached.

Test Conditions:

The simulator is initialized in the reference 100% power condition and the malfunctions activated. After saturation is reached, safety injection will be actuated to demonstrate recovery. The final conditions will be SI actuated with pressure, subcooling and vessel level recovering.

Date of Test:

January 21, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Simulator Test Certification Panel.

Test Results:

The test was performed successfully with some differences from the procedure observed.

Corrective Actions (if necessary):

The differences were evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.7, *Manual Reactor Trip*

Description of Test:

The test demonstrates the capability of the simulator to simulate a manual reactor trip. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

No malfunctions are used for this test. The reactor trip is initiated from a main control board switch.

The plant response will be a turbine trip and rapid cooldown to no-load temperature. Pressurizer pressure and level will decrease with the RCS cooldown. Feed flow to the steam generators will be isolated and the Emergency Feedwater System starts. Steam generator levels will decrease to the no-load level.

Test Conditions:

The simulator is initialized at the 100% power beginning of life condition and a manual reactor trip inserted. Manual actions are taken to stabilize the plant at hot standby, per the Emergency Operating Procedures. The final condition is Hot Standby with all controls rods inserted.

Date of Test:

January 21, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures

2. Simulator Test Certification Panel

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.8, *Simultaneous Trip Of All Feedwater Pumps*

Description of Test:

The test demonstrates the capability of the simulator to simulate a simultaneous trip of all feedwater pumps per Appendix B of ANSI/ANS 3.5-1985. This test also performs the safety analysis transient of a loss of all feedwater. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction FWM-1, "Feedwater Pump Trip", is used for this test. All of the main feedwater pumps are tripped by activating all three of the malfunction options. Malfunction FWM-3C is also used to trip the turbine-driven emergency feedwater pump, a condition of the accident analysis.

The plant response will be a turbine trip caused by the feedwater pumps tripping. The turbine trip then causes a reactor trip. The emergency feedwater pumps are started by the feedwater pumps tripping. The plant then will cooldown and stabilize at Hot Standby conditions.

Test Conditions:

The simulator is initialized in the 100% power end of life condition and the steam dump system is disabled, a condition of the accident analysis. The loss of feedwater is then initiated. The final condition is with the reactor tripped, at Hot Standby conditions with decay heat removal through the steam generator safety valves.

Date of Test:

January 22, 1991

Base Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Simulator Test Certification Panel.

Test Results.

The test was completed satisfactorily with a few minor differences noted in the procedure.

Corrective Actions (if necessary):

These discrepancies were evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Transient Test IST-7.9, *Trip Of One Reactor Coolant Pump*

Description of Test:

This test demonstrates the capability of the simulator to simulate a trip of any single reactor coolant pump per Appendix B of ANSI/ANS 3.5-1985. The test also performs the safety analysis transient of a partial loss of forced reactor coolant flow. The event is initiated from full power conditions, and the simulator response is compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis. The brush recorder traces are evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction RCS-3, "Reactor Coolant Pump Trip", is used for this test. This malfunction allows the selection of any of the RCPs to be tripped.

The plant response will be a reactor trip and cooldown to Hot Standby conditions. Steam flow from the affected steam generator will decrease and the affected loop temperature will be lower than the operating loops.

Test Conditions:

The simulator is initialized in the 100% beginning of life condition and the A Reactor Coolant Pump is tripped. The final conditions are Hot Standby with the reactor tripped.

Date of Test:

January 21, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15

2. Simulator Test Certification Panel

Test Results:

The test was completed satisfactorily with some differences from the procedure observed.

Corrective Actions (if necessary):

The differences were evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-7.10, *Maximum Rate Power Ramp (100% To 75% To 100%)*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a maximum rate power ramp per Appendix B of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response. Brush recorder traces are taken to compare the simulator response to the design response as shown in the setpoint study.

Malfunction Description:

No malfunctions are used for this test. Manual actions are only taken to change the turbine load set and to withdraw the control rods fully when returning to 100%.

Test Conditions:

The initial conditions are at the 100% middle of life condition. The power ramp is performed and the final condition is back at 100%.

Date of Test:

January 22, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station General Operating Procedures.
2. Setpoint Study for Virgil C. Summer Nuclear Station WCAP-9399.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Action (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Plant LER Test IST-10.1, *Reactor Trip Following Loss Of Inverter XIT-5904*.

Description of Test:

This test performs a comparison with a transient that occurred at the Virgil C. Summer Nuclear Station per Appendix A of ANSI/ANS 3.5-1985. The test duplicates a reactor trip that occurred due to the failure of Instrument Bus Inverter #4. The failure caused the loss of numerous instruments, among them the steam flow, feed flow, and SG level reference (from NI-44) inputs to the Steam Generator Water Level Control System. The SG feedwater regulating valves went shut and the reactor tripped on low SG level approximately one minute after the failure. The SG level trip setpoints were set to trip at a higher SG level at the time of this transient (1987) than they are at present.

Malfunction Description:

Malfunction EPS-11, "Loss of 120 VAC Instrument Bus", is used for this test. Any of the eight instrument buses can be selected to fail with this malfunction. Instrument bus 4 is the option selected to duplicated the plant event.

Test Conditions:

The simulator is initialized in the reference 100% power condition and conditions established for the test. The instrument bus is failed and after approximately 50 seconds manual FW control is attempted. The post-trip review printout is compared to the plant response for the period of time up to the reactor trip as a different reactor trip cause will be experienced. Final conditions for this test will be Hot Standby.

Date of Test:

March 8, 1991

Base Data References:

1. LER 87-0015, *Reactor Trip Package*.

Test Results:

The test was completed satisfactorily with good duplication of the plant pre-trip results.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Surveillance Test IST-10.2, *Reactor Trip After Loss of Power To I&C Process Control Cabinet XPN-7008.*

Description of Test:

This test performs a comparison to a transient experienced at the Virgil C. Summer Nuclear Station per Appendix A of ANSI/ANS 3.5-1985. The test duplicates a reactor trip event caused by the loss of power to I&C Process Cabinet XPN-7008. This cabinet powers the control loops for SG water level control, feedwater pump speed control, steam dump control, and pressurizer level and pressure control. Control of these processes was lost and a reactor trip occurred due to low SG level. The test sets up the same circumstances and verifies the same loss of control and reactor trip as occurred at the plant. A comparison of post-trip review printouts from the simulator and the plant provides the transient response comparison.

Malfunction Description:

Local operator action (LOA) EPS-176 is used for this test. This LOA allows the deenergizing of XPN-7008, the process control cabinet that failed.

Test Conditions:

The simulator is initialized in the reference 100% power condition. LOA EPS-176 is activated to start the transient. After the reactor trip, manual control of the SG power operated relief valves is taken to reduce RCS temperature. Power is restored to the cabinet at 13 minutes after initiation. The final conditions are at Hot Standby conditions with all control rods inserted.

Date of Test:

March 6, 1991

Baseline Data References:

1. LER 87-0027, *Reactor Trip Package.*

Test Results:

The test was completed satisfactorily with no problems observed. A very good correlation between piant and simulator response was achieved for this event.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Plant LER Test IST-10.3, *Safety Injection Actuation On Test Closure Of A MSIV*

Description of Test:

The test performs a transient that has been experienced at the Virgil C. Summer Nuclear Station per Appendix A of ANSI/ANS 3.5-1985. The transient performed is a safety injection actuation which occurred due to the inadvertent closing of a Main Steam Isolation Valve (MSIV). The MSIV closed during testing and resulted in a low steamline pressure SI actuation. This test duplicates the failure that occurred in the plant and compares the simulator response to the response of the plant. The post-trip review printout is used for this comparison.

Malfunction Description:

Malfunction MSS-6, "Main Steam Isolation Valve Failure", is used for this test. The malfunction has two options; an inadvertent closure, or a failure to close. An inadvertent closure with a two second ramp time is used for this test.

The closure of an MSIV at power causes the other two SGs to attempt to deliver the steam flow lost from the isolated SG. The steam pressure from the connected SGs drops rapidly and the low steamline pressure SI signal is actuated.

Test Conditions:

The simulator is initialized in the reference 100% power condition and the failure is initiated. The manual action of terminating SI, per the Emergency Operating Procedures, is taken at approximately seven minutes after initiation. The final conditions are at Hot Standby after terminating SI.

Date of Test:

January 23, 1991

Base Data References:

1. LER 88-0006, *Trip Package*

2. Virgil C. Summer Nuclear Station Emergency Operating Procedures.

Test Results:

The test was completed satisfactorily with no problems noted. Safety Injection was not terminated as called for by the procedure but the period of comparison only covered the time up to termination. Therefore, the results were not affected.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

. ..TIFICATION TEST ABSTRACT

Procedure Title:

Plant LER Test IST-10.5, *Reactor Trip Due To High Positive Power Rate.*

Description of Test:

The test performs a transient, that has been experienced at the Virgil C. Summer Nuclear Station per Appendix A of ANSI/ANS 3.5-1985. The transient performed is a reactor trip caused by a high rate of neutron flux. The event was caused by a miscalculation of the estimated critical position and the failure of the reactor operator to properly monitor the indications of criticality. This event occurred during core cycle #2. Although certain reactivity parameters changed with subsequent fuel loads, the event can still be duplicated producing the same approximate reactor core response. The simulator response will be compared to the LER summaries and printouts of the plant response contained in the trip package.

Malfunction Description:

N/A

Test Conditions:

The simulator is initialized in the Hot Standby Middle of Life condition and the RCS boron concentration is adjusted to achieve an estimated critical position at the highest integral rod worth (Bank D at approximately 40 steps). A reactor startup is conducted and, after the block of the source range hi flux reactor trip, the control rods are withdrawn continuously until a reactor trip occurs. The final condition is Hot Standby with all control rods inserted.

Date of Test:

February 7, 1991

Base Data References:

1. LER 85-0003, *Reactor Trip Package.*

Test Results:

The test was completed satisfactorily with no problems observ.d.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Plant LER Test IST-10.6, *Safety Injec..on Actuation Due To Pressurizer Spray Valve Failure.*

Description of Test:

This test performs a transient, that has been experienced at the Virgil C. Summer Nuclear Station, per Appendix A of ANSI/ANS 3.5-1985. The transient performed is a safety injection actuation due to low RCS pressure caused by the failing open of a pressurizer spray valve. This test duplicates the failure that occurred and the manual actions taken to mitigate the failure. The post-trip review printout of the event will be compared to the printout of the simulator response.

Malfunction Description:

Malfunction PRS-3, "Pressurizer Spray Valve Failure", is used for this test. The malfunction allows the selection of either spray valve to fail anywhere from 0% to 100% open.

Test Conditions:

The simulator is initialized at the reference 100% power condition and then power is reduced to 92%. Spray valve PCV-444D is failed to 100% open over 5 seconds. The reactor is manually tripped at 1980 PSIG and A Reactor Coolant Pump is manually tripped at approximately one minute thereafter. Manual actions, per the Emergency Operating Procedures, are taken to terminate safety injection at approximately eight minutes after the reactor trip. The final conditions are Hot Standby with all control rods inserted.

Date of Test:

January 31, 1991

Baseline Data References:

1. LER 85-0034, *Trip Package*.

2. Virgil C. Summer Nuclear Station Emergency Operating Procedures.

Test Results:

The test was performed with similar results to the plant event with the exception of an automatic Reactor Trip from overtemperature ΔT which occurred at an RCS Pressure of 2028 PSIG. This response was evaluated to be correct.

Corrective Actions(if necessary):

None.

Comments:

Changes in the OTΔT setpoint since 1985 increase the sensitivity of the OTΔT trip. The occurrence of the Reactor Trip at 2028 PSIG is a correct response with the present setpoint. (Ref: Lou Cartin TWR #22768'864).

3.4 Malfunction Test Abstracts

This section contains the test abstracts for the malfunction tests. The capability to simulate malfunctions is required in Section 3.1.2 of ANSI/ANS 3.5-1985.

All of the malfunction tests were completed satisfactorily except for IST-6.6.11.1, *Loss of 120 VAC Instrument Bus*. This test had some exceptions which are documented on Trouble Reports 372 and 445. These problems are scheduled to be corrected in the second quarter of 1991.

VIRGIL C. SUMMER SIMULATOR

C"RTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.1.1.9, *Loss Of Instrument Air.*

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of instrument air per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated from full power conditions and the simulator response is compared to the expected response.

Malfunction Description:

Malfunction AUX-1 "Loss of Instrument Air to All Headers", is used for this test. This malfunction simulates a break in one of the five auxiliary building headers, the reactor building header, the intermediate building header, or the turbine building header. In addition, a total loss of instrument air is simulated.

The response to a loss of instrument air depends upon the location selected. A total loss of instrument air will cause a reactor trip as major air-operated valves go to their failed positions.

Test Conditions:

The simulator is initialized at the reference 100% power condition. A total loss of instrument air is actuated and the response observed. The final condition will be with the reactor tripped and all simulated air-operated devices in their failed position.

Date of Test

January 15, 1991

Baseline Date References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with one difference from the procedure observed.

Corrective Actions: (if necessary)

The difference was determined to be a correct response, and a procedure change was submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.1.4.2, *Loss of Service Water System*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of service water per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the effect of a loss of service water is observed on the affected systems. The emergency diesel generators are operated in the emergency start condition to verify that they are cooled by the backup supply from the fire service system.

Malfunction Description:

Malfunction AUX-4, "Service Water Pump Trip", is used for this test. This malfunction provides the capability of tripping any or all of the service water pumps.

The response to a total loss of service water will be an increase in component cooling water system temperature, a trip of the HVAC chiller, and a trip of the emergency diesel generator, if it is running in test start condition. The diesel generators will receive backup cooling from the fire service system if running in emergency start. If service water is not restored, the plant will require eventual shutdown due to high temperature on the RCP motor bearings.

Test Conditions:

The simulator is initialized in the reference 100% power condition and all the service water pumps are tripped.

The test is stopped, still at full power, after the correct responses are verified.

Date of Test:

January 14, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily although some differences from the procedure were noted.

Corrective Actions: (if necessary)

The differences were evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.2.7.2, *Loss Of Component Cooling Water System*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of the component cooling system per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the effect of a loss of component cooling is observed on the affected systems and components.

Malfunction Description:

Malfunction CCW-7, "CCW Pump Trip", is used for this test. This malfunction provides the capability of tripping any or all of the component cooling water pumps.

A loss of component cooling will cause high temperatures in the supplied systems and components, with the Chemical and Volume Control System and the Reactor Coolant Pumps being the most critical during power operation.

Test Conditions:

The simulator is initialized at the reference 100% power condition and all the CCW pumps are tripped. The test is terminated after the indications of a loss of component cooling water have been verified.

Date of Test

January 11, 1991

Baseline Date References

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with one minor problem noted. One procedural error was noted due to the change in the TSC Computer System.

Corrective Actions: (if necessary)

Trouble Report #405 was written on the failure of the CCBP Discharge Pressure Low annunciator to alarm. The problem was corrected and tested satisfactorily on January 14, 1991. A procedure change was initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

<u>Procedure Title</u>:

Malfunction Test IST-6.3.1, *Loss Of Condenser Vacuum*.

<u>Description of Test</u>:

This test demonstrates the capability of the simulator to simulate a loss of the condenser vacuum per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions with the trip of one condenser vacuum pump under conditions of high condenser air in-leakage. Manual actions are taken, per the Abnormal Operating Procedures, to start a standby vacuum pump. After the effectiveness of that action is demonstrated, all condenser vacuum pumps are stopped, and the simulator response is compared to the expected response.

<u>Malfunction Description</u>:

Malfunction CND-1, "Main Condenser Vacuum Pump Trip", is used for this test. The malfunction simulates an oil leak or plugged filter on any of the three main condenser vacuum pumps.

Under normal conditions, the loss of one condenser vacuum pump will not cause an immediate problem. For this test, Local Operator Action CND-37 is used to simulate a high condenser air in-leakage rate. Condenser vacuum decreases causing a decrease in generator load, a turbine trip / reactor trip, and a block of steam dumps to the condenser.

<u>Test Conditions</u>:

The simulator is initialized at the reference 100% power conditions, and one of the main condenser vacuum pumps is selected to trip. After the verifications and manual actions are performed, the final conditions will be with the plant in Hot Standby.

<u>Date of Test:</u>

January 12, 1991

<u>Baseline Data References</u>

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted, except for one minor procedure error.

Corrective Actions: (if necessary)

A procedure change initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.3.2.1, *Hotwell Level Controller Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate loss of condenser level control per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at 75% power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction CND-2, "Hotwell Level Controller Failure", is used for this test. The malfunction provides the capability of failing either level controller LC-3001 or LC-3011 to the minimum or maximum level.

LC-3001 controls the normal condenser makeup valve and the condensate reject valve. LC-3011 controls the emergency condenser makeup valve. Failing one of the level controllers will not cause a condition leading to an immediate emergency condition. The condition tested, however, will cause a control action from each of the level controllers.

Test Conditions:

The simulator is initialized in the 75% power condition and level controller LC-3001 is failed high. The simulator will stabilize at the same power level with the condenser hotwell level lower than the initial level.

Date of Test:

January 12, 1991

Baseline Data References

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted. During the test, it was noted that one annunciator window had been omitted from the procedure.

Corrective Actions: (if necessary)

A procedure change was initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.4.1, *Rods Fail To Move*.

Description of Test:

This test demonstrates the capability of the simulator to simulate an inability to drive control rods per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions, and the inability to drive control rods in all modes is demonstrated. The condition is then cleared, and normal operation of the control rods is established.

Malfunction Description:

Malfunction CRF-1, "Rods Fail to Move", is used for this test. This malfunction simulates a failure in the rod control system logic cabinet which will prevent operation of any of the control rods.

The plant response is the inability to step any of the control rods in any mode or direction. The control rods are free to drop upon opening the reactor trip breakers.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the malfunction activated. After the verifications are performed, the malfunction is cleared and the simulator final conditions are at approximately 100% power.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed successfully with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.4.4.1, *Dropped Rod*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a rod drop per Section 3.1.2 of ANSI/ANS 3.5-1985. This test also duplicates a test described in FSAR Section 15.2.3. The event is initiated at full power conditions and the simulator response compared to the expected response. The dropped rod is then recovered as directed by the Abnormal Operating Procedure.

Malfunction Description:

Malfunction CRF-4, "Dropped Rod", is used for this test. This malfunction has two subsets so that any two of the rods may be selected. The failure simulated can be on the movable gripper coil or the stationary gripper coil.

A movable gripper coil failure will only be evidenced while the rod is being stepped. A stationary gripper coil failure will cause the rod to fall to the fully inserted position. The plant response will be a decrease in reactor power, a decrease in coolant temperature, and a decrease in pressurizer pressure and level. A flux tilt may be seen on the ex-core nuclear instrumentation depending upon the location of the dropped rod.

Test Conditions:

The simulator is initialized at the reference 100% power condition and one rod selected to trip due to a stationary coil failure. After the verification of alarms and indications, the dropped rod is recovered so the final conditions are back at 100% power.

Date of Test:

January 23, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6. 4.5, *Rod Ejection*.

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of a control rod ejection. The event is initiated at critical conditions and the simulator response compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves of this accident.

An attempt is made to match the accident analysis conditions, but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction CRF-5, "Ejected Rod", is used for this test. Any control rod may be selected for ejection and a leak size from 0 to 2000 gallons per minutes may be selected.

The plant response to an ejected rod during a reactor startup will be an increase in neutron flux and startup rate. The RCS leak caused by the rod ejection will cause a decrease in pressurizer pressure and level and a possible safety injection actuation.

Test Conditions:

The simulator is initialized in the Hot Standby End of Life condition, and a reactor startup is conducted. With the reactor critical, rod F6 is selected with a 1000 GPM leak size. The final conditions are with the reactor tripped and safety injection actuated.

Date of Test:

January 25, 1991

Baseline Date References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Simulator Test Certification Panel.

Test Results:

Although some differences from the procedure were noted, the test was completed satisfactorily.

Corrective Actions (if necessary):

The differences were evaluated and a procedure change was initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.4.6.1, *Uncontrolled Auto Rod Motion*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a failure in an automatic control system controlling reactivity per Section 3.1.2 of ANSI/ANS 3.5-1985. This event is initiated at full power conditions and the simulator response compared to the expected response. Manual action is taken to stop the transient, per the Abnormal Operating Procedures.

Malfunction Description:

Malfunction CRF-6, "Uncontrolled Rod Motion", is used for this test. This malfunction has two options available: uncontrolled auto rod motion and uncontrolled manual rod motion. Uncontrolled auto rod motion has a selectable rod speed of 8 to 72 steps per minute.

The plant response to uncontrolled auto rod motion is the continuous stepping of the control rods when they receive a motion signal. The rods will step at the speed selected for the malfunction. This action, assuming rods are inserting, will cause reactor power to decrease, T_{avg} to decrease, and pressurizer pressure and level to decrease. The manual action of placing rods in the Manual Mode, per the Abnormal Operating Procedures, will stop the transient.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the malfunction, set for uncontrolled auto motion at 72 steps per minute, is activated. After verification of the simulator response, the rods are placed in manual which terminates the transient at approximately 90% to 100% power.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Te[illegible]fication Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.4.6.2, *Uncontrolled Manual Rod Motion*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of uncontrolled RCCA bank withdrawal. The event is initiated from a "just critical" condition, and the simulator response is compared to the expected response. Brush recorder traces are taken to compare the simulator response to the safety analysis curves for this accident.

An attempt is made to match the accident analysis conditions, but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction CRF-6, "Uncontrolled Rod Motion", is used for this test. This malfunction has two options available: uncontrolled automatic rod motion or uncontrolled manual rod motion. The uncontrolled automatic rod motion option has a selectable rod speed of from 8 to 72 steps per minute.

The plant response to an uncontrolled manual rod withdrawal will be increasing neutron flux and startup rate. If this occurs while operating at power, RCS temperature will also increase.

Test Conditions:

The simulator is initialized in the Hot Standby condition and the reactor is taken critical. The malfunction is initiated, uncontrolled manual motion at 48 steps per minute. The final conditions will be Hot Standby with all rods on the bottom after a reactor trip.

Date of Test:

January 17, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.4.7.2, *Stuck Rod*

Description of Test:

This test demonstrates the capability of the simulator to simulate a stuck rod and a misaligned rod per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the inability to step or to trip the stuck rod is demonstrated.

Malfunction Description:

Malfunction CRF-7, "Stuck Rod", is used for this test. This malfunction has two subsets so that any two of the rods may be selected. The failure mode offers the options of causing the rod to be trippable or untrippable.

The plant response to a stuck rod will be a rod misalignment alarm when the rod is misaligned from its respective group. The selected rod will remain out on a reactor trip.

Test Conditions:

The simulator is initialized at the reference 100% power condition and rod F10 is selected to be untrippable. The indications and alarms are verified during rod stepping and the reactor is manually tripped. The final condition is in hot standby with rod F10 stuck out.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.5.7, *Loss Of Normal Letdown*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a failure of the Reactor Coolant Volume Control System per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response. Manual actions are taken, as directed by the Abnormal Operating Procedures, to control pressurizer level.

Malfunction Description:

Malfunction CVC-7, "Loss of Normal Letdown", is used for this test. This malfunction simulates the loss of output signal from LB-459C.

The response to this failure is a trip of the pressurizer heaters, a closure of letdown isolation valve LCV-459, and the closure of the letdown orifice isolation valves. Pressurizer level will increase due to the loss of letdown flow and pressurizer pressure will slowly decrease due to the loss of heaters.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the malfunction activated. The proper responses are verified, the malfunction is cleared, and a normal 100% lineup is established.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with some minor differences from the procedure.

Corrective Actions: (if necessary)

The differences were evaluated and a procedure change initiated.

Comments:

The Simulator Test Certification Panel will review this test at its next meeting.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.5.12, *Leak In Charging Line*.

Description of Test:

This test demonstrates the simulator capability to simulate the emergency event of loss of coolant inside containment per Section 3.1.2 of ANSI/ANS 3.5-1985. The leak is initiated at full power conditions and the simulator response compared to the expected response. Manual actions are then taken to isolate the leak.

Malfunction Description:

Malfunction CVC-12, "Leak in Charging Line", is used for this test. This malfunction simulates a leak on the charging line inside containment between the non-regenerative heat exchanger and the connections to the RCS. The leak size is selectable from 0 to 100% of the charging flow and can be ramped over a period of up to 3600 seconds.

The response to a charging line leak will be high charging line flow, decreasing pressurizer level, increase in RB sump inventory, and a decreased letdown temperature.

Test Conditions:

The simulator is initialized in the reference 100% power condition. A 100% charging line leak is initiated and the proper indications and alarms are verified. Manual actions are then taken to isolate the charging line leak.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.5.13, *Letdown Line Leak Outside Containment*.

Description of Test:

This test demonstrates the simulator capability to simulate the emergency event of a loss of coolant outside containment per Section 3.1.2 of ANSI/ANS 3.5-1985. The leak is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction CVC-13, "Letdown Line Leak Outside Containment", is used for this test. Malfunction CVC-13 simulates a leak downstream of PCV-145 with a selectable leak size of 0 to 100% of the letdown flow. The leak can be ramped over a period of up to 3600 seconds.

The response to a letdown line leak is a decrease in the volume control tank level and increased makeup.

Test Conditions:

The simulator is initialized in the reference 100% power condition and the 95% letdown line leak is initiated. The leak will be isolated by operator action.

Date of Test:

January 25, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was performed initially with a response different from that specified in the procedure. The Nuclear Station has been operating with a letdown temperature that is less than the design temperature, therefore, the Vital Area Temperature Monitor will not cause an isolation of the letdown line on high temperature.

Corrective Actions: (if necessary)

A procedure change was submitted to reflect this change in response.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.1, *Station Blackout*

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of offsite power per Section 3.1.2 of ANSI/ANS 3.5-1985. In addition, this test duplicates the FSAR test of a loss of offsite power to the station auxiliaries. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction EPS-1, "Station Blackout", is used for this test. Malfunction EPS-1 simulates a sequential tripping of offsite power sources due to weather conditions.

The malfunction sequence starts with a loss of the emergency auxiliary transformer which deenergizes ESF Bus 1DB. The unit auxiliary transformer is then lost, which deenergizes the service buses and causes a reactor trip. The ESF transformers are lost which then deenergizes ESF Bus 1DA. The emergency diesel generators auto-start and energize their respective ESF buses, and the safety-related equipment is started by the ESF load sequencers.

Test Conditions:

The simulator is initialized in the reference 100% power and condition and the station blackout initiated. The simulator response is observed to the final condition of the RCS at no-load conditions in natural circulation with the ESF buses and equipment powered by their respective emergency diesel generators.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.4.1, *Loss Of Service Bus*

Description of Test:

This test demonstrates the capability of the simulator to simulate the loss of power to the plant's electrical distribution buses per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction EPS-4, "Loss of Service Bus", is used for this test. This malfunction simulates an overcurrent trip condition on any one of the three 7.2 KV service buses.

The plant response to this event will be a loss of power to the selected 7.2 KV service bus, its associated 480 V buses and load centers, and the BOP equipment served by these buses. A reactor trip will occur, if greater than 38% power, due to the loss of one reactor coolant pump.

Test Conditions:

The simulator is initialized in the reference 10. % power condition and the malfunction initiated on the A service bus. Th ndications of loss of power to the bus and its associated loads are verified as w as a reactor trip due to the trip of the A Reactor Coolant Pump. The final conditions are at Hot Standby with two reactor coolant pumps running.

Date of Test:

January 11, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.6.1, *Diesel Generator Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate the loss of emergency generators per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction EPS-6, "Diesel Generator A Failure", is used for this test. The malfunction simulates a mechanical failure of the emergency diesel generator with the capability to select either or both of the diesel generators.

The response to this malfunction will be that, when called upon to start, the selected diesel generator will fail. Under emergency start conditions, this results in the loss of power to the respective ESF Bus and its associated loads.

Test Conditions:

The simulator is initialized in the reference 100% power condition and a failure of A DG is selected . The normal supply breaker to ESF Bus 1DA is opened and the correct indications of the loss of power are verified. The final conditions are at 100% power with ESF Bus 1DA deenergized.

Date of Test:

January 16, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with the exception of one omission from the procedure and a minor problem with the Radiation Monitoring System response.

Corrective Actions: (if necessary)

Trouble Report #408 was written to correct RMS meter responses on a loss of power. This problem was corrected and tested satisfactorily on March 2, 1991. A procedure change was also submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.7.1, *Loss Of 125 VDC Bus*

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of power to DC instrumentation buses per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction EPS-7, "Loss of 125 VDC Bus 1HA", is used for this test. Malfunction EPS-7 simulates a ground on 125 VDC Bus 1HA (A Train), 1HB (B Train), or 1HX (non-train). The simulated ground causes a trip of the respective battery breaker and battery charger, which deenergizes the selected bus and its related loads.

Loss of a 125 VDC bus will cause a reactor trip, loss of MCB control of affected equipment, and loss of various indicators and annunciators.

Test Conditions:

The simulator is initialized in the reference 100% power condition and the station blackout initiated. The simulator response is observed to the final condition of the RCS at no-load conditions in natural circulation and the ESF buses and equipment powered by their respective emergency diesel generators.

Date of Test:

January 22, 1991

Baseline Data References

1. General Maintenance Procedure GMP-112.000, *Electrical Feeder List*.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with several minor problems noted and with some differences from the procedure.

Corrective Actions: (if necessary)

Simulator Discrepancy Reports (SDR) 91009 and 91010 were written on the simulator problems discovered. Both SDRs were corrected and tested satisfactorily on January 22, 1991. The procedure differences were evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.11.1, *Loss of 120 VAC Instrument Bus*

Description of Test:

This test demonstrates the capability of the simulator to simulate the loss of power to AC instrumentation buses per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction EPS-11, "Loss of 125 VAC Instrument Bus APN-5901", is used for this test. The malfunction simulates the failure of one of the eight 120 VAC inverters supplying power to NSS and BOP instrument buses.

The loss of a 120 VAC instrument bus will cause the loss of the instrumentation, indication, or control devices associated with that bus. The bus loss will not cause a reactor trip, except for Instrument Bus #6, but on the NSS buses a partial reactor trip condition will occur.

Test Conditions:

The simulator is initialized in the reference 100% power condition and the malfunction activated on Instrument Bus #1. The indications of loss of power are verified. The final conditions are 100% power with a partial reactor trip from the failed protection channels.

Date of Test:

March 2, 1991

Baseline Data References:

1. General Maintenance Procedure GMP-112.000, *Electrical Feeder List*.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with several minor problems, in that there were some errors and omissions in the procedure.

Corrective Actions: (if necessary)

Trouble Reports #372 and #445 were written. A procedure change was also initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.12, *Generator Breaker Trips Without Turbine Trip*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a generator trip per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response

Malfunction Description:

Malfunction EPS-12, "Generator Breaker Trips Without Turbine Trip", is used for this test. This malfunction simulates a failure causing the generator breaker to trip without a turbine trip.

The response to this event will be a turbine overspeed and a large power mismatch. The turbine control system will close the control and intercept valves to limit the overspeed condition. Tavg will rapidly increase and a Reactor Trip will occur if at full power.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the malfunction activated.

The final conditions are at hot standby after a Reactor Trip due to the rapid RCS temperature increase (Overtemperature ΔT Trip)

Date of Test:

January 12, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station General Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was initially performed with results considerably different than those specified by the procedure.

Corrective Actions: (if necessary)

The simulator response was evaluated to be correct and a procedure change was submitted to revise the procedure.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.6.18, *Failure Of ESF Transformer.*

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of emergency power per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction EPS-18, "Failure of ESF Transformer", is used for this test. The malfunction simulates an overcurrent condition which causes a transformer lockout of either or both ESF transformers.

ESF Transformer XTF-4 is the normal source of ESF power. Failure of transformer XTF-4 (Option 1) will cause a trip of the transformer's circuit switcher (high side supply) and the normal supply breaker to ESF Bus 1DA. If Bus 1DB was powered from the alternate source, it's supply breaker would also trip. Under normal conditions, Bus 1DA would be deenergized and Emergency Diesel Generator A would emergency start and re-energize the bus. If ESF Transformer XTF-5 was energized and selected as the source of ESF power, then a failure of transformer XTF-5 (Option 2) would cause a trip of the transformer's circuit switcher, the ESF Transformer tie breaker, and the normal supply breaker to ESF Bus 1DA. The Emergency Diesel Generator A would emergency start and re-energize the bus.

Test Conditions:

The simulator is initialized in the reference 100% power condition and one of the transformer failures is actuated. The final conditions will still be at 100% power with ESF Bus 1DA powered by DG A if XTF-4 was the selected failure.

Date of Test:

December 21, 1990

Baseline Data References:

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

Due to recent changes to the electrical distribution system, both ESF transformer failure models were tested.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.7.1.1, *Main Feedwater Pump Trip*

Description of Test:

This test demonstrates the capability of the simulator to simulate a feedwater system failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction FWM-1, "Main Feedwater Pump Trip", is used for this test. This malfunction allows the tripping of any or all of the main feedwater pumps.

The trip of one feedwater pump at full power will cause a decrease in feedwater flow and steam generator levels. The remaining two feedwater pumps will increase in speed, thus making up for the capacity lost from the tripped pump. The steam generator levels will be restored and operation can continue, though the operating feedwater pumps will be providing their rated capacity.

Test Conditions:

The simulator is initialized at the reference 100% power condition and one of the feedwater pumps is tripped. The test is terminated when plant conditions have stabilized and the steam generator levels are restored.

Date of Test:

January 12, 1991

Baseline Data References

1. Simulator Test Certification Panel.
2. Virgil C. Summer Nuclear Station Annunciator Response Procedures.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.7.1.2, *Loss Of Normal And Emergency Feedwater.*

Description of Test:

This test demonstrates the capability of the simulator to simulate the loss of all feedwater per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response is compared to the expected response. After the reactor is tripped, manual actions are taken per the Emergency Operating Procedures to establish core cooling using an RCS bleed and feed.

Malfunction Description:

Malfunction FWM-1, "Main Feedwater Pump Trip", and FWM-3, "Emergency Feedwater Pump Trip", are used for this test. Each malfunction allows for the tripping of any or all of their respective feedwater pumps.

The plant response initially will be decreasing steam generator levels as the main feedwater pumps trip. A reactor trip occurs on low steam generator levels or when the last feedwater pump trips and trips the turbine. The motor-driven emergency feedwater pump does not start and the turbine-driven pump starts on Lo-Lo levels in two steam generator and then trips. The EOPs then direct manual actions to attempt to add feedwater to the steam generators. When the SG water inventory is depleted, the EOPs direct manual actions to establish core cooling by RCS bleed and feed.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the loss of all feedwater activated. After the manual actions directed by the EOPs, the final conditions will be hot standby with core cooling provided by an RCS bleed and feed.

Date of Test:

January 19, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
2. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.7.3.2, *Turbine Driven Emergency Feedwater Pump Trip*

Description of Test:

This test demonstrates the capability of the simulator to simulate a passive malfunction in the emergency feedwater system per Section 3.1.2 of ANSI/ANS 3.5-1985. The failure is activated with the pump shutdown, where it is not apparent. A manual start is then attempted and the indications of a pump trip are verified.

Malfunction Description:

Malfunction FWM-3, "Emergency Feedwater Pump Trip", is used for this malfunction. This malfunction will trip either of the motor-driven pumps or the turbine-driven pump.

The plant response to a turbine-driven emergency feedwater pump trip is a decrease in the emergency feedwater flow to the steam generators and a decrease in steam generator levels.

Test Conditions:

The simulator is initialized in the 2% power condition and the failure activated. A automatic start is attempted and the failure indications verified. The final conditions are still at 2% power.

Date of Test

January 19, 1991

Baseline Date References

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.7.8, *Feedline Break Inside Containment.*

Description of Test:

This test demonstrates the capability of the simulator to simulate a main feed line break inside containment per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response. Manual actions are taken, per the Emergency Operating Procedures, to isolate the affected steam generator.

Malfunction Description:

Malfunction FWM-8, "Feedline Break Inside Containment", is used for this test. This malfunction simulates an unisolable break on any of the feedlines inside containment. The break size is selectable from 0-12E6 LB/HR and can be ramped in at up to 3600 seconds.

The plant response to a feedline break will be decreasing steam generator level and increasing feedwater flow on the affected steam generator. A reactor trip will occur due to the steam generator level. The containment temperature and pressure will increase eventually causing a safety injection. Plant conditions will start to stabilize after the affected steam generator is isolated and blows dry.

Test Conditions:

The simulator is initialized at the reference 100% power condition and one steam generator selected for the malfunction. A break size of two million pounds per hour, ramped in over 60 seconds, is selected. The final conditions will be with the reactor tripped, safety injection actuated, and the affected steam generator isolated and empty.

Date of Test

January 12, 1991

Baseline Dat> References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with some differences from the procedure observed.

Corrective Actions: (if necessary)

The differences were evaluated as correct response and a procedure change was submitted.

Comments:

The Simulator Test Certification Panel will review this test at its next meeting.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.7.15.1, *FW Control Valve Position Failure (Excessive Heat Removal).*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of excessive heat removal due to a feedwater system malfunction. The event is initiated from full power conditions and the simulator response compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves of this accident.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction FWM-15, "Feedwater Control Valve Position Failure", is used for this test. This malfunction allows the selection of any FW control valve to fail in any position.

The plant response to this event will be an increasing SG level, decreasing RCS temperature, and increasing nuclear power. The transient will be terminated by a reactor trip on high level in the affected steam generator.

Test Conditions:

The simulator is initialized in the reference 100% power condition and one FW control valve is selected to fail to 100% open in 2 seconds. The final condition will be Hot Standby with the reactor tripped.

Date of Test:

January 14, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with some differences from the procedure observed.

Corrective Actions (if necessary):

The differences were evaluated as correct responsc and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.7.16.1, *Feedline Break Between FE-496 and FW ISOL VLV*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a main feedline break outside containment per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction FWM-16, "Feedline Break Outside Containment", is used for this test. This malfunction has four selectable locations available; on the C SG feedline, on the B SG feedline, on the combined FW header before the feed regulating valves, or on the discharge of the A feedwater pump. The break size can be selected from 0 to 12E6 LB/HR, ramped over up to 3600 seconds.

The plant response to this event will be decreased feed flow to the SG(s) and decreasing levels. A reactor trip on SG level will be experienced for large leak sizes.

Test Conditions:

The simulator is initialized in the reference 100% power condition and Malfunction FWM-16, Break on C SG feedline between FE-496 and FW ISOL VLV, is selected. A break size of 6E6 LB/HR, ramped over 180 seconds, is selected. The final conditions will be Hot Standby with the steam generator levels being recovered by emergency feedwater.

Date of Test

January 12, 1991

Baseline Date References

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.8.3, *Steamline Break Inside Containment*

Description of Test:

This test demonstrates the capability of the simulator to simulate a main steamline break inside containment per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response. Manual actions, as directed by the Emergency Operating Procedures, are taken to isolate the affected steam generator.

Malfunction Description:

Malfunction MSS-3, "Steamline Break Inside Containment", is used for this test. This malfunction allow the selection of any of the three steam generators to experience a selectable size leak or break (0 to 12E6 LB/HR). The leak can be ramped in at up to 3600 seconds.

The plant response to a large steamline break will be a rapid cooldown and depressurization. The containment temperature and pressure will also rapidly increase. A reactor trip and safety injection will occur. The event is terminated when affected steam generator is isolated and blows dry.

Test Conditions:

The simulator is initialized at the reference 100% power condition and one steam generator is selected. A leak size of 2E6 LB/HR is selected to ramp in over 60 seconds. Manual actions are taken to isolate the affected steam generator per the Emergency Operating Procedures. The final conditions are shutdown with safety injection actuated and the affected steam generator isolated and empty.

Date of Test:

January 12, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no probl ms noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.8.4, *Steamline Break Outside Containment*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a main steamline break outside containment per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response. Manual actions, as directed by the Emergency Operating Procedures, are taken to isolate the affected steam generator.

Malfunction Description:

Malfunction MSS-4, "Steamline Break Outside Containment", is used for this test. This malfunction simulates a break on the selected steamline at a location outside the containment but upstream of the main steam isolation valve. The break size is selectable between 0 to 12E6 LB/HR and can be ramped in at up to 3600 seconds.

The plant response to a large steamline break outside containment will be rapidly decreasing RCS temperature and pressure. Reactor power will increase until the reactor is automatically tripped. Manual actions are taken to isolate the affected steam generator per the Emergency Operating Procedures. The affected steam generator blows dry, which actuates safety injection but stops the RCS cooldown.

Test Conditions:

The simulator is initialized at the reference 100% power condition and one steam generator is selected to experience the break. A leak size of six million pounds per hour, ramped over 180 seconds, is selected. The final conditions will be with all rods inserted, safety injection actuated, and the selected steam generator isolated and empty.

Date of Test

February 8, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.

2. Simulator Test Certification Panel.

Test Results:

The test was completed successfully with some differences from the procedure observed.

Corrective Actions: (if necessary)

The differences were evaluated as correct responses and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.8.5.2, *Steamline Dump Control Failure*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a failure of an automatic control system controlling core heat removal. The event is initiated at lower power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction MSS-5, "Steam Dump Control Failure", is used for this test. This malfunction provides the capability of failing the steam dump control signal anywhere from 0% to 100%. The high failure (100%) would open all the steam dump valves if armed, while the low failure (0%) would prevent any of the steam dump valves from modulating open.

The plant response, to a low failure would be closure of the steam dump valves, an increase in T_{avg}, and an increase in steamline pressure. The SG power-operated relief valves would open, when steam pressure reached their setpoint, providing an alternate means of core cooling.

Test Conditions:

The simulator is initialized at the 2% power condition and the malfunction, set to 0%, is activated. The final conditions will be at approximately 2% power with heat removal through the SG power-operated relief valves.

Date of Test:

January 14, 1991

Baseline Data References

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.8.10, *SG Safety Valve Failure (MS Accidental Depressurization)*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of a main steam system accidental depressurization. The event is initiated from Hot Standby conditions and the simulator response compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves of this accident.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction MSS-10, "Steam Generator Safety Valve Failure", is used for this test. This malfunction allows the selection of a SG safety valve from any of the steam generator to be failed from 0% to 100% open.

The plant response to this event from Hot Standby will be decreasing RCS temperature, decreasing pressurizer level, and decreasing pressurizer pressure. A reactor trip and Safety Injection will occur on low Pressurizer Pressure. The transient is terminated when the affected steam generator boils dry after it is isolated per the Emergency Operating Procedures.

Test Conditions:

The simulator is initialized in the Hot Standby EOL condition. One of the SG safety valves is selected to fail at 100% open. The final conditions with safety injection actuated, and the affected steam generator isolated and empty.

Date of Test

January 17, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily though some differences from the procedure were noted.

Corrective Actions: (if necessary):

The differences were evaluated as correct responses and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.8.10, *SG Safety Valve Failure (MS Accidental Depressurization)*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of a Main Steam System accidental depressurization. The event is initiated from Hot Standby conditions and the simulator response compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves of this accident.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction MSS-10, "Steam Generator Safety Valve Failure", is used for this test. This malfunction allows the selection of a SG safety valve from any of the steam generator to be failed from 0% to 100% open.

The plant response to this event from Hot Standby will be decreasing RCS temperature, decreasing pressurizer level, and decreasing pressurizer pressure. A reactor trip and Safety Injection will occur on low Pressurizer Pressure. The transient is terminated when the affected steam generator boils dry after it is isolated per the Emergency Operating Procedures.

Test Conditions:

The simulator is initialized in the Hot Standby EOL condition. One of the SG safety valves is selected to fail at 100% open. The final conditions with safety injection actuated, and the affected steam generator isolated and empty.

Date of Test

January 17, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily though some differences from the procedure were noted.

Corrective Actions: (if necessary)

The differences were evaluated as correct responses and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.9.3.2, *Power Range Control Channel Failure*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a nuclear instrumentation failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated from 75% power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction NIS-3, "Power Range Channel Failure", is used for this test. This malfunction simulates the failure of the summing and level amplifier of any of the four power range channels. The failed output can be set from 0 to 200% power and ramped at up to 3600 seconds.

The plant response to a low failure of a power range channel would be the actuation of the reactor trip bistable associated with the failed channel and numerous alarms. The high failure would the actuation of the reactor trip bistable associated with the failed channel and numerous alarms. The output of power range channel IV (NI-44) also provides input to the rod control system and the Steam Generator Water Level Control System. A low failure of NI-44 would cause the control rods to step out and the steam generator control level to decrease to the no-load value.

Test Conditions:

The simulator is initialized in the 75% power condition and Channel IV (NI-44) is failed to 0%..

The final conditions will be at approximately 75% power with the control rods withdrawn to 220 steps and the steam generator levels being controlled at no-load level.

Date of Test

January 11, 1991

Baseline Date References

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily, with the exception of a hardware problem with recorder NR-47. This problem did not affect the test results.

Corrective Actions: (if necessary)

Simulator Discrepancy Report 90-114 was written to repair the faulty recorder.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

<u>Procedure Title</u>:

Malfunction Test IST-6.10.3, *Steam Generator Level Control Failure*.

<u>Description of Test</u>:

This test demonstrates the capability of the simulator to simulate a process control system failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared is to the expected response.

<u>Malfunction Description</u>:

Malfunction PCS-3, "Steam Generator Level Control Failure", is used for this test. This malfunction simulates a steam generator level transmitter failure on the transmitter that supplies input to the Steam Generator Water Level Control System. A failure can be selected on any steam generator with a failure mode of anywhere from 0% to 200%, with the 100% setting equal to "fails as is".

The plant response to a low failure would be to increase feed flow to the affected steam generator which will raise the level. A high failure will cause a decrease in feed flow which will reduce the steam generator level. Either failure will result in a reactor trip unless manual actions are taken.

<u>Test Conditions</u>:

The simulator is initialized in the reference 100% power condition. One level transmitter is selected to fail high (200%) over 10 seconds. The final conditions will be in Hot Standby with the reactor tripped.

<u>Date of Test</u>

January 12, 1991

<u>Baseline Data References</u>

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.10.5.1, *Safety Injection Failure (Inadvertent Actuation).*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of an inadvertent actuation of the ECCS system at power. The event is initiated at full power conditions and the simulator response compared to the expected response.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction PCS-5, "Safety Injection Failure (Inadvertent Initiation)", is used for this test. This malfunction allows either or both safety injection trains to fail to actuate or to inadvertently actuate.

The plant response to an inadvertent ECCS actuation will be a reactor trip/turbine trip and a sequenced start of the ECCS equipment. After the initial cooldown and drop in pressurizer level and pressure from the reactor trip, ECCS flow will start increasing pressurizer level and pressure.

Test Conditions:

The simulator is initialized in the reference 100% power condition and both ECCS trains are inadvertently actuated. The final condition is hot standby with the reactor tripped and safety injection actuated.

Date of Test:

January 12, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactory with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.10.9.1, *Reactor Trip Breaker Failure (Inadvertent Open)*

Description of Test:

This test demonstrates the capability of the simulator to simulate a reactor trip per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at 10^{-3}% power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction PCS-9, "Reactor Trip Breaker Failure", is used for this test. This malfunction provides the capability of two failure modes for each trip breaker; inadvertent opening or failure to open. A failure to open can be from an automatic trip signal, a manual trip signal, or both.

The response to an inadvertent opening of one breaker at the test power level will be a drop of all control rods and a decrease in reactor power to shutdown level. The non-selected reactor trip breaker will remain closed.

Test Conditions:

The simulator is initialized at 10^{-3}% power and one reactor trip breaker is selected for inadvertent opening. The correct response is then verified. The final conditions for this test will be at Hot Standby wtih all rods inserted; the selected reactor trip breaker will be open and the other breaker closed.

Date of Test:

February 21, 1991

Baseline Date References:

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with one procedural discrepancy noted.

Corrective Actions: (if necessary)

A procedure change was initiated to correct the procedure.

Comments:

The Simulator Test Certification Panel reviewed and approved the procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.10.9.2, *Reactor Trip Breaker Failure (Fails To Open).*

Description of Test:

This test demonstrates the capability of the simulator to simulate a failure of the Automatic Reactor Trip System. The event is initiated at full power conditions and the simulator response compared to the expected response. A failure to trip from both automatic and manual signals is demonstrated.

Malfunction Description:

Malfunction F 9, "Reactor Trip Breaker Failure", is used for this test. Either or both of the reac r trip breakers can be selected to inadvertently open or fail to open. The failu o open can be initiated from an automatic trip signal, a manual trip signal, or bo...

The response to this failure will be a failure of the selected reactor trip breaker to open in response to the selected signal. A reactor trip will still occur as long as the other reactor trip breaker operates correctly.

Test Conditions:

The simulator is initialized in the reference 100% power condition and one reactor trip breaker is selected to fail to open from an automatic signal. This selection is then tested and verified. The simulator is initialized again at 100% power and the other breaker is selected to fail to open from both manual and automatic signals. This selection is tested and verified. The final conditions are in Hot Standby with the reactor tripped.

Date of Test

January 17, 1991

Baseline Date References

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.11.1.1, *Pressurizer Pressure Control Channel Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a failure of the Reactor Coolant Pressure Control System per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at 50% power to demonstrate all of the control actions without a reactor trip. The proper simulator response is verified and then manual action is taken, per the Abnormal Operating Procedures, to regain pressure control.

Malfunction Description:

Malfunction PRS-1, "Pressurizer Pressure Control Channel Failure (HI)", is used for this test. This malfunction allows the selection of pressure transmitter, PT-444 or PT-445 to fail at a selected pressure (anywhere from 1700 to 2500 PSIG). PT-444 provides the control signal for the pressurizer heaters, the spray valves, and one power-operated relief valve. PT-445 provides the control signal to the other two power-operated relief valves. A failure low on PT-445 causes no action, while on PT-444 it causes heaters to go full on and spray valves to close. This transient is fairly slow and therefore is not selected. A high failure of either channel will cause the opening of power-operated relief valve (s) and spray valves (PT-444). Pressurizer pressure will decrease rapidly, causing various alarms and a decrease in the Overtemperature differential temperature, (OTΔT), reactor trip setpoint. The P-11 interlock will close the power-operated relief valves when its setpoint is reached, stopping the pressure decrease. Manual actions are then taken to control pressure.

Test Conditions:

The simulator is initialized at 50% power and one of the pressure channels is selected to fail high (2500 PSIG). The proper response is verified and manual actions, per the Abnormal Operating Procedures, are taken to restore conditions to the initial conditions.

Date of Test:

January 15, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

Both options of the malfunction were tested satisfactorily with some difference from the procedure noted for PT-445.

Corrective Actions: (if necessary)

The differences were evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.11.4.2, *Relief Valve Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a pressurizer power operated relief valve failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction PRS-4, "Relief Valve Failure (Interlock not Functional)", is used for this test. This malfunction allows selection of either pressurizer power operated relief valve with the P-11 interlock functional or not functional. The selected valve can be failed to any position (0-100%) and over any time frame from 0 to 3600 seconds.

The plant response to a pressurizer power operated relief valve failure would be increased tailpipe temperature and a decrease in pressurizer pressure. If the P-11 interlock is functional, the relief valve will close when the pressurizer pressure decreases to 1985 PSIG. If the interlock is not functional, the pressurizer pressure decreases until Safety Injection actuation.

Test Conditions:

The simulator is initialized in the reference 100% power condition. One of the pressurizer power operated relief valves is selected to fail 50% open with the P-11 interlock not functional. The event proceeds with a reactor trip and safety injection actuation and is terminated when the safety injection flow has started to restore pressurizer level and pressure.

Date of Test:

January 15, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.11.5, *Pressurizer Pressure Channel Failure (Protection)*

Description of Test:

This test demonstrates the capability of the simulator to simulate the loss of a protective system channel per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated and the simulator response compared to the expected response.

Malfunction Description:

Malfunction PRS-5, "Pressurizer Pressure Channel Failure (Protection)", is used for this test. This malfunction allows the selection of any of the three pressurizer pressure protection channels and the selection of any failed pressure value between 1700 to 2500 PSIG.

The response to a pressurizer pressure channel failing low is a partial reactor trip and safety injection actuation signal from the selected channel, a partial reactor trip from the overtemperature differential temperature (OT Δ T) channel fed by selected pressure channel, and alarms on the Core Cooling Monitor fed by the selected pressure channel.

Test Conditions:

The simulator is initialized in the reference 100% power condition and a pressurizer pressure channel is selected to fail low. The partial trip indications and alarms are verified while the simulator remains at 100% power.

Date of Test:

January 15, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Abnormal Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.11.7, *Safety Valve Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a pressurizer safety valve failure per Section 3.1.2 of ANSI/ANS 3.5-1985. In addition, this test duplicates the FSAR event of an accidental depressurization of the RCS. The event is initiated at full power conditions and the simulator response compared to the expected response and to the accident analysis response.

Malfunction Description:

Malfunction PRS-7, "Safety Valve Failure", is used for this test. This malfunction has the capability to fail any one of the three pressurizer safety valves from 0% to 100% open, ramped in at up to 3600 seconds.

The indications caused by a safety valve failure would be an increase in safety valve tail pipe temperature and flow increase/alarm on the safety valve flow monitor. Pressurizer pressure would decrease causing a reactor trip and safety injection actuation.

Test Conditions:

The simulator is initialized in the reference 100% power condition and safety valve 8010B is failed 100% open. The event proceeds with a reactor trip and safety injection actuation, and is terminated when the safety injection flow has started to restore pressurizer level and pressure.

Date of Test:

January 15, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with one discrepancy. This concerned one annunciator window which had been removed at the plant under MCN 21465J.

Corrective Actions: (if necessary)

A procedure change was initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.2, *Steam Generator Tube Leak*

Description of Test:

This test demonstrates the simulator's capability to simulate emergency events per Section 3.1.2 of ANSI/ANS 3.5-1985. A design basis steam generator tube leak is initiated at full power conditions, and the simulator response is compared to the expected response. Manual actions, as directed by the Virgil C. Summer Nuclear Station Emergency Operating Procedures, are taken to mitigate the event and terminate the primary to secondary leakage.

Malfunction Description:

Malfunction RCS-2, "Steam Generator Tube Leak" is used for this test. This malfunction simulates the tube leakage of any steam generator from very slight leakage up to a total tube shear (0-600 GPM). The leakage can be ramped in over 0 to 3600 seconds.

The plant response for a large SG tube leak will be a decrease in RCS pressure and pressurizer level, and an increase in the secondary plant activity with subsequent secondary plant radiation alarms. A decrease in SG feedwater flow will be apparent on the affected SG. A reactor trip on overtemperature differential temperature or low pressurizier pressure will occur, followed by an automatic safety injection actuation.

Test Conditions:

The simulator is initialized in the reference 100% power condition, the affected steam generator is selected, and a maximum size (600 GPM) leak is started. Manual actions, per the Emergency Operating Procedures, are taken to terminate the primary to secondary leakage. The final conditions of this test are with RCS pressure and ruptured SG pressure equalized. Additionally, backfill from the ruptured SG to the RCS is demonstrated.

Date of Test:

January 16, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
3. Westinghouse Owners Group Emergency Response Guidelines, Background Documents.
4. North Anna Steam Generator Tube Rupture Event, INPO Case Study 88-014.
5. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.3, *Reactor Coolant Pump Trip.*

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of forced core coolant flow per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at a power level below the single loop loss of flow reactor trip setpoint, and the simulator response is compared to the expected response.

Malfunction Description:

Malfunction RCS-3, "Reactor Coolant Pump Trip", is used for this test. This malfunction provides the capability of tripping any of the reactor coolant pumps.

The trip of a reactor coolant pump usually results in a reactor trip unless the reactor power level is below the P-8 setpoint, 38% power.

Test Conditions:

The simulator is initialized in the 25% power condition and one of the RCPs is selected to be tripped. The simulator response is verified as conditions are stabilized at approximately 25% power with two RCS loops in service.

Date of Test:

January 15, 1991

Baseline Data References

1. Simulator Test Certification Panel

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.12.4, *Reactor Coolant Pump Locked Rotor.*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of a reactor coolant pump locked rotor. The event is initiated at full power conditions, and the simulator response is compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves of this accident.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction RCS-4, "Reactor Coolant Pump Locked Rotor", is used for this test. This malfunction allows for the selection of any reactor coolant pump to experience a locked rotor.

The plant response to a locked rotor will be an immediate stoppage of flow in the selected loop. RCS temperature, pressurizer level, and pressurizer pressure will increase until the reactor is tripped. The pressurizer power operated relief valves or safety valves may actuate. Steam flow from the affected loop steam generator will stop.

Test Conditions:

The simulator is initialized in the reference 100% power condition and B RCP is selected. The final conditions are Hot Standby with all rods on the bottom.

Date of Test:

February 15, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.5, *Large LOCA In Cold Leg*

Description of Test:

This test demonstrates the simulator capability to simulate a large reactor coolant break per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions, and the simulator response is compared to the expected response. Manual actions, as directed by the Emergency Operating Procedures, are taken to achieve the cold leg recirculation lineups.

Malfunction Description:

Malfunction RCS-5, "Large LOCA In Cold Leg", is used for this test. This malfunction simulates a cold leg break on a selectable RCS Loop. The break size is a Design Basis Accident shear.

The plant response for any break size is a rapid RCS depressurization and an automatic safety injection actuation.

Test Conditions:

The simulator is initialized in the reference 100% power condition. A DBA LOCA is initiated on Loop A. Manual actions, per the EOPs, are taken to change the safety injection system alignment from injection to cold leg recirculation. The final conditions are cold leg recirculation with the reactor core covered.

Date of Test:

January 16, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures
2. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
3. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with one difference from the procedure observed.

Corrective Actions: (if necessary)

The difference was evaluated and a procedure change submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.6.1, *Reactor Coolant System Leak*

Description of Test:

This test demonstrates the capability of the simulator to simulate a Reactor Coolant System leak per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at 75% power and the simulator response compared to the expected response.

Malfunction Description:

Malfunction RCS-6, "Reactor Coolant System Leak", is used for this test. This malfunction provides a selectable 0-20,000 GPM leak on any selected RCS loop. The leak can be ramped in over up to 3600 seconds.

The response to an RCS leak is a decrease in RCS pressure and pressurizer level. A leak greater than makeup capability will cause a reactor trip and safety injection actuation. Indications of leakage into the containment will also be seen.

Test Conditions:

The simulator is initialized in the 75% power condition. A 175 GPM leak, ramped in over 180 seconds, is activated. The manual action of starting a second charging pump is performed to increase makeup capability. The final condition is with letdown isolated, RCS makeup flow matching leak flow, and suction to the charging pumps from the RWST.

Date of Test:

January 25, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.12.6.2, *Small Break LOCA*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of a small break LOCA. The event is initiated from full power conditions, and the simulator response is compared to the expected response. Brush recorder traces are taken to compare the simulator response to the accident analysis curves for this accident.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction RCS-6, "Reactor Coolant System Leak", is used for this test. The malfunction allows the selection of any loop to experience a leak of 0 to 20,000 gallons per minute.

The small break LOCA response will be a decreasing pressurizer pressure and lev and an eventual safety injection actuation. The RCS water inventory and pressu will continue to decrease until the safety injection accumulators inject. RCS water inventory will then start to recover.

Test Conditions:

The simulator is initialized in the 100% power EOL condition. An RCS leak of 11,500 GPM on A Loop is initiated. The final conditions are with safety injection actuated, RCS pressure at SI accumulator pressure, and core subcooling and reactor vessel level increasing.

Date of Test:

January 25, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily though a difference in response from that specified in the procedure was noted.

Corrective Actions (if necessary):

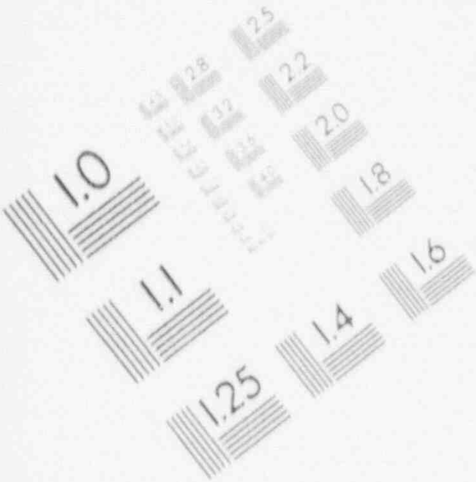
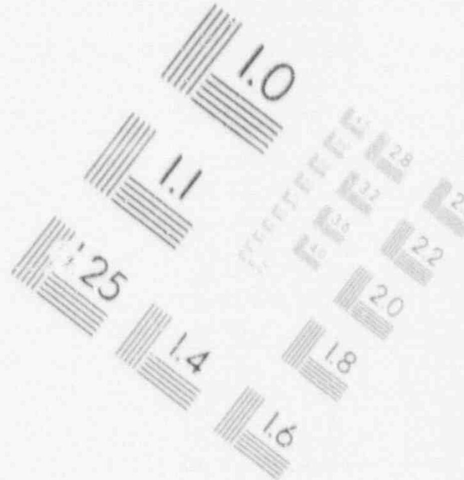
The difference was evaluated and a procedure change was submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

2

IMAGE EVALUATION TEST TARGET (MT-3)

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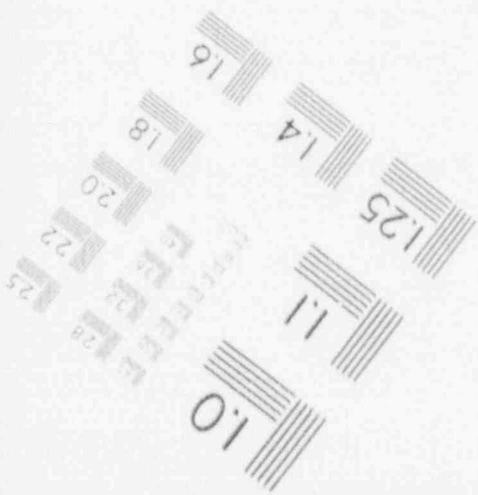
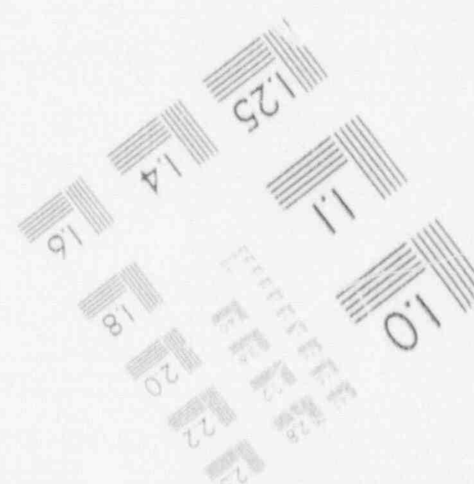

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WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.8.1, *RCS Hot Leg RTD Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a process instrumentation failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at
100% power and the simulator response compared to the expected response.

Malfunction Description:

Malfunction RCS-8, "Reactor Coolant System RTD Failure", is used for this test. This malfunction allows the selection of any of the nine hot leg RTDs to fail anywhere between 530°F and 630°F.

The effect of failing a hot leg RTD will be a change in that loop's Tavg and Δ T. The hot leg RTDs for each loop are averaged so the amount of change will be relatively small but MCB annunciators and a partial Reactor Trip may be received.

Test Conditions:

The simulator is initialized in the 100% power condition, and a selected hot leg RTD is failed to 630° F. The final condition will be at 100% power after the simulator response is verified.

Date of Test:

January 23, 1991

Baseline Data References

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily although some differences from the procedures were noted.

Corrective Actions: (if necessary)

These differences were evaluated and a procedure change was initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.8.2, *RCS Cold Leg RTD Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a process instrumentation failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at
100% power and the simulator response compared to the expected response.

Malfunction Description:

Malfunction RCS-8, "Reactor Coolant System RTD Failure", is used for this test. This malfunction allows the selection of any of the three cold leg RTDs to fail anywhere between 530°F and 630°F.

The effect of failing a cold leg RTD will be a change in that loop's Tavg and ΔT. MCB annunciators will be activated, and a partial Reactor Trip may be received.

Test Conditions:

The simulator is initialized in the 100% power condition and a selected hot leg RTD is failed to 630° F. The final condition will be at 100% power after the simulator response is verified.

Date of Test:

January 23, 1991

Baseline Data References

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily although some differences from the procedures were noted.

Corrective Actions: (if necessary)

The differances were evaluated and a procedure change was submitted.

Comments:

The Simulator Test Certification Panel reviewed and approved the test results and procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.9.1, *RCS Median Signal Selector Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a process instrumentation failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at 100% power and the simulator response compared to the expected response.

Malfunction Description:

Malfunction RCS-9, "RCS Median Signal (Tavg or ΔT Selector Failure)", is used for this test. This malfunction provides the capability of failing the signal from the Tavg Median Signal Selector or the ΔT Median Signal Selector to the maximum or minimum value.

Failure of the ΔT Median Signal Selector will actuate only MCB annunciators. Failure of the Tavg Median Signal Selector will cause the programmed pressurizer level to change, the control signal to the Steam Dump System to change, and the control rods to step.

Test Conditions:

The simulator is initialized in the 100% power condition, and the Tavg Median Signal Selector is failed to 630° F. The simulator is stabilized at the same power level after the proper responses are verified.

Date of Test:

January 25, 1991

Baseline Data References:

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with one omitted indicating light noted on the procedure.

Corrective Actions: (if necessary)

A procedure change was initiated.

Comments:

The Simulator Test Certification Panel reviewed and approved the procedure change.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.10, *RCS Fuel Leak*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a fuel cladding failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction RCS-10, "Reactor Coolant System Fuel Leak", is used for this test. This malfunction simulates a fuel clad failure causing RCS activity to increase to the selectable range of 10^{-2} uc/cc to 10 uc/cc. This activity can be ramped in at up to 3600 seconds.

The response to this failure is various radiation monitoring system alarms.

Test Conditions:

The simulator is initialized at the reference 100% power condition and the malfunction activated and set at 10 uc/cc ramped in over 60 seconds. The test is terminated, still at 100% power, after verification of the correct responses.

Date of Test:

December 4, 1990

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.12.15, *RCS Flow Transmitter Failure*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a process instrumentation and alarm failure per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at full power conditions, and the simulator response is compared to the expected response.

Malfunction Description:

Malfunction RCS-15, "Reactor Coolant System Flow Transmitter Failure", is used for this test. This malfunction allow the selection of any of six of the loop flow transmitters to fail at from 0% to 120%.

The plant response to any single flow transmitter failure would be a partial reactor trip and associated alarms. The failure of two transmitters low on the same RCS loop would cause a reactor or trip.

Test Conditions:

The simulator is initialized in the reference 100% power condition. One of the flow transmitters is selected to fail to 0% over 10 seconds. After verification of the correct responses, the final condition is still at 100% power.

Date of Test:

November 9, 1990

Baseline Data References:

1. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

FSAR Test IST-6.12.18, *Variable RCS Boron Concentration (Uncontrolled Dilution).*

Description of Test:

This test demonstrates the capability of the simulator to simulate the safety analysis accident of an uncontrolled boron dilution. The event is initiated from Hot Standby conditions, and the simulator response is compared to the expected response.

An attempt is made to match the accident analysis conditions but all of the conservative assumptions used cannot be duplicated. The simulator response is therefore evaluated under the criteria that the simulator parameters change in the same direction as the accident analysis and do not violate physical laws of nature.

Malfunction Description:

Malfunction RCS-18, "Variable RCS Boron Concentration", is used for this test. This malfunction allows the setting of the RCS boron concentration at any value, ramped in at up to 3600 seconds.

The plant response to an uncontrolled dilution will be an increased neutron flux and eventual criticality.

Test Conditions:

The simulator is initialized in the Hot Standby condition and the malfunction activated. The final condition is with the reactor critical from the dilution.

Date of Test:

January 18, 1991

Baseline Data References:

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report, Chapter 15.
2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems noted.

Corrective Actions (if necessary):

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.13.1.2, *Loss Of Shutdown Cooling*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a loss of shutdown cooling per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at conditions of approximately 190 °F, and 400 PSIG, and the effect of a loss of shutdown cooling observed on the Reactor Coolant System.

Malfunction Description:

Malfunction RHR-1, "RHR Pump Trip", is used for this test. The malfunction allows for the trip of either or both of the RHR pumps.

The response to a loss of shutdown cooling will be a gradual heatup of the reactor core and reactor coolant system. This will be seen by increases in RCS Loop T_{HOT} and pressurizer level.

Test Conditions:

The simulator is initialized at RCS conditions of approximately 190°F and 400 PSIG. All RHR pumps are then tripped. The test is terminated when the indications of an RCS heatup are observed.

Date of Test:

January 18, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.

2. Simulator Test Certification Panel.

Test Results:

The test was completed satisfactorily with no problems observed.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER TRAINING SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.13.7, *Reactor Building Spray Pump Failure*

Description of Test:

This test demonstrates the capability of the simulator to simulate a passive malfunction in a ESF system per Section 3.1.2 of ANSI/ANS 3.5-1985. The failure is initiated at full power conditions, where it is not apparent. A loss of coolant accident is then activated, and the indications of a reactor building spray pump failure are verified.

Malfunction Description:

Malfunction RHR-7, "Reactor Building Spray Pump Failure," is used for this test. This malfunction allows the selection of either reactor building spray pump to trip or fail to start.

The activation of a pump failure while the pump is not running provides no indication of the failure. The selected pump will fail to start if activated, or trip if running, causing a loss of flow and pressure and a pump trip annunciator.

Test Conditions:

The simulator is initialized in the reference 100% power condition, and one spray pump is selected to trip/fail to start. A DBA LOCA (Malfunction RCS-5) is activated and the indications of the pump failure are verified. The final conditions are in the injection phase of a loss of coolant accident with the selected reactor building spray pump not running.

Date of Test

January 18, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Virgil C. Summer Nuclear Station Annunciator Response Procedures.
3. Simulator Test Certification Panel.

Test Results:

This test was completed satisfactorily with no problems noted.

Corrective Actions: (if necessary)

None.

Comments:

None.

VIRGIL C. SUMMER SIMULATOR

CERTIFICATION TEST ABSTRACT

Procedure Title:

Malfunction Test IST-6.15.1, *Inadvertent Turbine Trip*.

Description of Test:

This test demonstrates the capability of the simulator to simulate a turbine trip per Section 3.1.2 of ANSI/ANS 3.5-1985. The event is initiated at a full power conditions and the simulator response compared to the expected response.

Malfunction Description:

Malfunction TUR-1, "Inadvertent Turbine Trip", is used for this test. This malfunction simulates a mechanical trip valve failure which trips the turbine.

The plant response to a turbine trip at greater than 50% power will be a reactor trip and cooldown to hot standby conditions.

Test Conditions:

The simulator is initialized at the reference 100% condition and the malfunction activated. The final condition will be at hot standby conditions, after verification of the simulator response.

Date of Test:

January 18, 1991

Baseline Data References

1. Virgil C. Summer Nuclear Station Emergency Operating Procedures.
2. Simulator Test Certification Panel.

Test Re[illegible]s:

The test was completed satisfactorily with one minor problem with the servo valve current indications

Corrective Actions: (if necessary)

Simulator Discrepancy Report 91014 was written to correct the servo valve current indication.

Comments:

None.

3.5 Simulator Annual Operability Testing Requirements

Simulator performance testing is required on a continuous basis to verify overall simulator model completeness and integration, and to verify simulator performance against the performance criteria of ANSI/ANS 3.5-1985.

A simulator annual operability test is required by the standard, and is comprised of steady state performance tests and transient tests of a benchmark set of transients. In addition, the simulator real time test will be performed annually.

Regulatory Guide 1.149, *Nuclear Power Plant Simulation Facilities For Use in Operator License Examinations*, requires that all the certified simulator malfunctions be tested in their entirety over a four year period. This is in addition to the required annual operability test.

Appendix 1 lists the simulator testing requirements over the four year period.

Simulator Annual Operability Test Requirements

1991/1992

A. Steady State Performance Tests (Annual)

1) IST-4.1, 100 Percent Power Heat Balance-Stability Test.

2) IST-4.2, 75 Percent Power Heat Balance.

3) IST-4.3, 50 Percent Power Heat Balance.

B. Transient Tests (Annual)

1) IST-7.1, Simultaneous Closure of All 3 MSIVs.

2) IST-7.2, Simultaneous Trip of All 3 RCPs.

3) IST-7.3, Main Turbine Trip (At Maximum Power Level Which Does Not Cause Reactor Trip).

4) IST-7.4, 100% LOCA With Loss of Offsite Power.

5) IST-7.5, 100% Unisolable Main Steamline Break.

6) IST-7.6, Slow Primary System Depressurization to Saturated Conditions.

7) IST-7.7, Manual Reactor Trip.

8) IST-7.8, Simultaneous Trip of All Feedwater Pumps.

9) IST-7.9, Trip of One Reactor Coolant Pump.

10) IST-7.10, Maximum Rate Power Ramp (100% to 75% to 100%).

C. Computer Real Time Test (Annual)

1) IST-11.1, Computer Real Time Test.

D. Normal Operations Tests (25% per Year)

1) IST-2.1, Power Reduction to 75 Percent Power.

2) IST-2.2, Power Reduction to 50 Percent Power.

3) IST-2.3, 50 Percent Power to 2 Percent Power.

E. Malfunction Tests (25% per Year)

1) IST-6.12.2, Steam Generator Tube Leak.
2) IST-6.12.6.1, Reactor Coolant System Leak.
3) IST-6.6..1, Station Blackout.
4) IST-6.6.11.1, Loss of 120 VAC Instrument Bus.
5) IST-6.3.2.1, Hotwell Level Controller Failure.
6) IST-6.7.1.1, Main Feedwater Pump Trip.
7) IST-6.4.4.1, Dropped Rod.
8) IST-6.6.12, Generator Breaker Trips Without Turbine Trip.
9) IST-6.5.7, Loss of Normal Letdown.
10) IST-6.7.8, Feedline Break Inside Containment.
11) IST-6.10.3, Steam Generator Level Control Failure.
12) IST-6.4.6.2, Uncontrolled Manual Rod Motion.
13) IST-6.10.5.1, Safety Injection Failure (Inadvertent Initiation).
14) IST-6.12.8.1, Reactor Coolant System Hot Leg RTD Failure.

F. Plant LER Tests (25% per Year)

1) IST-10.1, Reactor Trip Following Loss of Inverter XIT-5904.
2) IST-10.5, Reactor Trip Due To High Positive Power Rate.

1992/1993

A. Steady State Performance Tests (Annual)

1) IST-4.1, 100 Percent Power Heat Balance-Stability Test.

2) IST-4.2, 75 Percent Power Heat Balance.

3) IST-4.3, 50 Percent Power Heat Balance.

B. Transient Tests (Annual)

1) IST-7.1, Simultaneous Closure of All 3 MSIVs.

2) IST-7.2, Simultaneous Trip of All 3 RCPs.

3) IST-7.3, Main Turbine Trip (At Maximum Power Level Which Does Not Cause Reactor Trip).

4) IST-7.4, 100% LOCA With Loss of Offsite Power.

5) IST-7.5, 100% Unisolable Main Steamline Break.

6) IST-7.6, Slow Primary System Depressurization to Saturated Conditions.

7) IST-7.7, Manual Reactor Trip.

8) IST-7.8, Simultaneous Trip of All Feedwater Pumps.

9) IST-7.9, Trip of One Reactor Coolant Pump.

10) IST-7.10, Maximum Rate Power Ramp (100% to 75% to 100%).

C. Computer Real Time Test (Annual)

1) IST-11.1, Computer Real Time Test.

D. Normal Operations Tests (25% per Year)

1) IST-2.4, 2 Percent Power to Hot Standby.

2) IST-2.5, Hot Standby to Hot Shutdown.

3) IST-2.6, Shutdown and Cooldown to Cold Shutdown.

E. Malfunction Tests (25% per Year)

1) IST-6.5.12, Leak in Charging Line.

2) IST-6.11.7, Safety Valve Failure.

3) IST-6.6.18, Failure of ESF Transformer.

4) IST-6.6.7.1, Loss of 125 VDC Bus.

5) IST-6.1.4.2, Loss of Service Water System.

6) IST-6.7.1.2, Loss of Normal and Emergency Feedwater.

7) IST-6.4.1, Rods Fail to Move.

8) IST-6.4.6.1, Uncontrolled Auto Rod Motion.

9) IST-6.10.9.1, Rx Trip Breaker Failure (Inadvertent Open).

10) IST-6.7.16.1, Feedline Break Between FE-496 and FW Isolation Valve

11) IST-6.13.7, Reactor Building Spray Pump Failure.

12) IST-6.4.5, Ejected Rod.

13) IST-6.8.10, Steam Generator Safety Valve Failure.

14) IST-6.12.8.2, Reactor Coolant System Cold Leg RTD Failure.

F. Plant LER Tests (25% per year)

1) IST-10.2, Reactor Trip After Loss of Power to I&C Process Control Cabinet XPN-7008.

1993/1994

A. Steady State Performance Tests (Annual)

1) IST-4.1, 100 Percent Power Heat Balance-Stability Test.

2) IST-4.2, 75 Percent Power Heat Balance.

3) IST-4.3, 50 Percent Power Heat Balance.

B. Transient Tests (Annual)

1) IST-7.1, Simultaneous Closure of All 3 MSIVs.

2) IST-7.2, Simultaneous Trip of All 3 RCPs.

3) IST-7.3, Main Turbine Trip (At Maximum Power Level Which Does Not Cause Reactor Trip).

4) IST-7.4, 100% LOCA With Loss of Offsite Power.

5) IST-7.5, 100% Unisolable Main Steamline Break.

6) IST-7.6, Slow Primary System Depressurization to Saturated Conditions.

7) IST-7.7, Manual Reactor Trip.

8) IST-7.8, Simultaneous Trip of All Feedwater Pumps.

9) IST-7.9, Trip of One Reactor Coolant Pump.

10) IST-7.10, Maximum Rate Power Ramp (100% to 75% to 100%).

C. Computer Real Time Test (Annual)

1) IST-11.1, Computer Real Time Test.

D. Normal Operations Tests (25% per Year)

1) IST-2.7, Cold Shutdown to Hot Shutdown.

2) IST-2.8, Hot Shutdown to Hot Standby.

E. Malfunction Tests (25% per Year)

1) IST-6.5.13, Letdown Line Leak Outside Containment.
2) IST-6.11.4.2, Relief Valve Failure.
3) IST-6.6.6.1, Diesel Generator Failure.
4) IST-6.12.3, Reactor Coolant Pump Trip.
5) IST-6.13.1.2, Loss of Shutdown Cooling.
6) IST-6.11.5, Pressurizer Pressure Channel Failure (Protection).
7) IST-6.12.10, RCS Fuel Leak.
8) IST-6.8.5.2, Steam Dump Control Failure.
9) IST-6.8.3, Steamline Break Inside Containment.
10) IST-6.9.3.2, Power Range Control Channel Failure.
11) IST-6.7.3.2, Turbine Driven Emergency Feedwater Pump Trip.
12) IST-6.7.15.1, FW Control Valve Position Failure.
13) IST-6.12.4, Reactor Coolant Pump Locked Rotor.
14) IST-6.12.9.1, Reactor Coolant System Tavg Median Signal Selector.

F. Plant LER Tests (25% per Year)

1) IST-10.3, Safety Injection Actuation on Test Closure of A MSIV.

1994/1995

A. Steady State Performance Tests (Annual)

1) IST-4.1, 100 Percent Power Heat Balance-Stability Test.
2) IST-4.2, 75 Percent Power Heat Balance.
3) IST-4.3, 50 Percent Power Heat Balance.

B. Transient Tests (Annual)

1) IST-7.1, Simultaneous Closure of All 3 MSIVs.
2) IST-7.2, Simultaneous Trip of All 3 RCPs.
3) IST-7.3, Main Turbine Trip (At Maximum Power Level Which Does Not Cause Reactor Trip).
4) IST-7.4, 100% LOCA With Loss of Offsite Power.
5) IST-7.5, 100% Unisolable Main Steamline Break.
6) IST-7.6, Slow Primary System Depressurization to Saturated Conditions.
7) IST-7.7, Manual Reactor Trip.
8) IST-7.8, Simultaneous Trip of All Feedwater Pumps.
9) IST-7.9, Trip of One Reactor Coolant Pump.
10) IST-7.10, Maximum Rate Power Ramp (100% to 75% to 100%).

C. Computer Real Time Test (Annual)

1) IST-11.1, Computer Real Time Test.

D. Normal Operations Tests (25% per Year)

1) IST-2.9, Hot Standby to 2 Percent Power.
2) IST-2.10, 2 Percent Power to 100 Percent Power.
3) IST-2.11, Recovery to Rated Power after Reactor Trip..

E. <u>Malfunction Tests</u> (25% per Year)

1) IST-6.12.5, Large LOCA in Cold Leg.
2) IST-6.1.1.9, Loss of Instrument Air.
3) IST-6.6.4.1, Loss of Service Bus.
4) IST-6.3.1, Loss of Condenser Vacuum.
5) IST-6.2.7.2, Loss of Component Cooling Water System.
6) IST-6.4.7.2, Stuck Rod.
7) IST-6.15.1, Inadvertent Turbine Trip.
8) IST-6.11.1.1, Pressurizer Pressure Control Channel Failure.
9) IST-6.8.4, Steamline Break Outside Containment.
10) IST-6.12.15, RCS Flow Transmitter Failure.
11) IST-6.10.9.2, Rx Trip Breaker Failure (Fails to Open).
12) IST-6.12.6.2, Small Break LOCA.
13) IST-6.12.18, Variable RCS Boron Concentration.

F. <u>Plant LER Tests</u> (25% per Year)

1) IST-10.6, Safety Injection Actuation Due to Pressurizer Spray Valve Failure.

4.0 SIMULATOR DISCREPANCY RESOLUTION AND UPGRADING

4.1 Discrepancy Resolution Procedure

An aggressive program of identifying and correcting simulator discrepancies (differences in the physical or operational characteristics between the simulator and the plant) has been in place since the start of training on the Virgil C. Summer Simulator. The program allows anyone noting a difference between the simulator and the Virgil C. Summer Nuclear Station to originate a Simulator Discrepancy Report (SDR). SDRs are also written to incorporate plant modifications into the simulator and to track temporary control board changes.

Simulator Discrepancy Reports are sent to Nuclear Computer Services for resolution. After the required hardware or software changes are made and tested by Nuclear Computer Services, the Simulator Operations Specialist will also test the changes and accept or reject the problem resolution. Only after satisfactory testing will a permanent change be made on the training disk. The status of all SDRs is maintained on the simulator Configuration Management System (CMS).

Administrative control of Simulator Discrepancy Reports is provided by Nuclear Operations Education and Training.

4.2 Upgrading Program

All modifications to the Virgil C. Summer Nuclear Station are reviewed by Nuclear Operations Education and Training for applicability to the simulator. Those modifications identified as requiring a simulator hardware or software change are documented with a Simulator Discrepancy Report. The status of all plant modifications, whether implemented on the simulator or not, are tracked by the Nuclear Operations Education and Training Department. The review and disposition of plant modifications are administratively controlled by Nuclear Operations Education and Training.

In addition, transients that have occurred at the Virgil C. Summer Nuclear Station and other industry events are reviewed by Nuclear Operations Education & Training for their training impact. A Simulator Discrepancy Report will be written to add new simulator malfunctions if deemed necessary for training. This program is administratively controlled by Nuclear Operations Education & Training.

5.0 LIST OF STANDARD ABBREVIATIONS

AB	-	Auxiliary Building
AC	-	Alternating Current
AFD	-	Axial Flux Difference
ALMS	-	Acoustic Leak Monitoring System
ANS-3.5	-	ANSI/ANS 3.5-1985
AO	-	Auxiliary Operator
AOP	-	Abnormal Operating Procedure
ARP	-	Annunciator Response Procedure
ATWS	-	Anticipated Transient Without a Scram
BAT	-	Boric Acid Tank
BCMS	-	Boron Concentration Monitoring System
BOL	-	Beginning of Life
BOP	-	Balance of Plant
BRS	-	Boron Recycle System
BS	-	Bistable
BTRS	-	Boron Thermal Regeneration System
CB	-	Control Building
CCBP	-	Component Cooling Water Booster Pump
CCW	-	Component Cooling Water
CCWP	-	Component Cooling Water Pump
CRDM	-	Control Rod Drive Mechanism
CREP	-	Control Room Evacuation Panel
CRS	-	Control Room Supervisor
CRT	-	Cathode Ray Tube
CS	-	Containment Spray
CSAS	-	Containment Spray Actuation Signal
CSF	-	Critical Safety Function
CST	-	Condensate Storage Tank
CVCS	-	Chemical and Volume Control System

D/G or DG	-	Diesel Generator
DAST	-	Deaerator Storage Tank
DBA	-	Design Basis Accident
DBD	-	Design Basis Document
DC	-	Direct Current
DMIMS	-	Digital Metal Impact Monitoring System
DNB	-	Departure From Nucleate Boiling
DNBR	-	Departure from Nucleate Boiling Ratio
DRPI	-	Digital Rod Position Indication
ECCS	-	Emergency Core Cooling System
ECP		Estimated Critical Position
EFW	-	Emergency Feedwater
EHC	-	Electrohydraulic Control
EOF	-	Emergency Offsite Facility
EOL	-	End of Life
EOP	-	Emergency Operating Procedure
EPP	-	Emergency Plan Procedure
ESF	-	Engineered Safety Features
ESFAS	-	Engineered Safety Features Actuation System
FCV	-	Flow Control Valve
FE	-	Flow Element
FHB	-	Fuel Handling Building
FSAR	-	Final Safety Analysis Report
FT	-	Flow Transmitter
FWBV	-	Feedwater Regulatng Bypass Valve
FWIV	-	Feedwater Isolation Valve
FWRV	-	Feedwater Regulating Valve
GOP	-	General Operating Procedure
gpm	-	Gallons Per Minute

Hz	- Hertz
I&C	- Instrumentation and Controls
IA	- Instrument Air
IB	- Intermediate Building
INPO	- Institute of Nuclear Power Operations
IPCS	- Integrated Plant Computer System
IR	- Intermediate Range
ISOL	- Isolation
KHz	- Kilohertz
KVA	- Kilovolt-Amps
LB	- Level Bistable
LBS/HR	- Pounds Mass Per Hour
LCO	- Limiting Condition for Operation
LCV	- Level Control Valve
LER	- Licensee Event Report
LOA	- Local Operator Action
LOCA	- Loss of Coolant Accident
LS	- Level Switch
MCB	- Main Control Board
MDEFP	- Motor Driven Emergency Feed Pump
MFBP	- Main Feedwater Booster Pump
MFP	- Main Feedwater Pump
MFW	- Main Feedwater
MI	- Miscellaneous Instrumentation
MOL	- Middle of Life
MOV	- Motor Operator Valve
MS	- Main Steam
MSIV	- Main Steam Isolation Valve
MTC	- Moderator Temperature Coefficient
MW	- Megawatt

MWe	-	Megawatt Electric
MWt	-	Megawatt Thermal
NIS	-	Nuclear Instrumentation System
NPSH	-	Net Positive Suction Head
NRC	-	Nuclear Regulatory Commission
NROTC	-	Nuclear Reactor Operator at The Controls
NSSS	-	Nuclear Steam Supply System
ONO	-	Off-Normal Occurrence
OPΔT	-	Overpower Differential Temperature
OTΔT	-	Overtemperature Differential Temperature
PCV	-	Pressure Control Valve
POAH	-	Point of Adding Heat
PORV	-	Power Operated Relief Valve (Pzr)
PR	-	Power Range
PRT	-	Pressurizer Relief Tank
PS	-	Pressure Switch
PSIG	-	Pounds Per Square Inch Gauge
PWR	-	Pressurized Water Reactor
PZR	-	Pressurizer
RB	-	Reactor Building
RCCA	-	Rod Cluster Control Assembly
RCDT	-	Reactor Coolant Drain Tank
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RHR	-	Residual Heat Removal
RIL	-	Rod Insertion Limit
RMWST	-	Reactor Makeup Water Storage Tank
RO	-	Reactor Operator
RPI	-	Rod Position Indication
RPM	-	Revolutions Per Minute

RPS	-	Reactor Protection System
RTB	-	Reactor Trip Breaker
RTD	-	Resistance Temperature Detector
RVLIS	-	Reactor Vessel Level Indicating System
RWST	-	Refueling Water Storage Tank
S/G or SG	-	Steam Generator
SDM	-	Shutdown Margin
SE	-	Shift Engineer
SER	-	Significant Event Report
SFP	-	Spent Fuel Pool
SFPCS	-	Spent Fuel Cooling System
SGTR	-	Steam Generator Tube Rupture
SGWLC	-	Steam Generator Water Level Control
SI	-	Safety Injection
SIPCS	-	Simulator Integrated Plant Computer System
SIS	-	Safety Injection System
SOER	-	Significant Operating Experience Report
SOP	-	System Operating Procedure
SR	-	Source Range
SRO	-	Senior Reactor Operator
SS	-	Shift Supervisor
STA	-	Shift Technical Advisor
STP	-	Surveillance Test Procedure
SUR	-	Startup Rate
SW	-	Service Water
SWBP	-	Service Water Booster Pump
SWP	-	Service Water Pump
TAVG or Tavg or T_{avg}	-	Average Temperature
TB	-	Turbine Building

T_C	-	Cold Leg Temperature
TC	-	Thermocouple
TDEFP	-	Turbine Driven Emergency Feed Pump
T_H	-	Hot Leg Temperature
TREF	-	Reference Temperature
TS	-	Technical Specification
TSC	-	Technical Support Center
V	-	Volt
VAC	-	Volts-Alternating Current
VAR	-	Volt-Ampere Reactive
VCT	-	Volume Control Tank
VDC	-	Volts-Direct Current
VLV	-	Valve
WGDT	-	Waste Gas Decay Tank
WG	-	Waste Gas System
ΔP	-	Differential Pressure
ΔT	-	Differential Temperature