UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

July 23, 2002

Virginia Electric and Power Company
ATTN: Mr. David A. Christian
Sr. Vice President and
Chief Nuclear Officer
Innsbrook Technical Center - 2SW
500 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY NUCLEAR POWER STATION - NRC INSPECTION REPORT 50-280/02-07, 50-281/02-07

Dear Mr. Christian:

On June 27, 2002, the Nuclear Regulatory Commission (NRC) completed a safety system design and performance capability inspection at your Surry Nuclear Power Station. The enclosed report documents the inspection findings which were discussed on June 27, 2002, with Mr. Bryan Foster and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your operating license. Within these areas, the inspection involved selected examination of procedures and representative records, observations of activities, and interviews with personnel.

No findings of significance were identified during the inspection.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

D. Charles Payne, Acting Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-280, 50-281
License Nos.:DPR-32, DPR-37

Enclosure: (See page 2)

Enclosure: NRC Inspection Report 50-280/02-07
50-281/02-07 w/Attachment

cc w/encl:
Stephen P. Sarver, Director
Nuclear Licensing and
Operations Support
Virginia Electric & Power Company
Electronic Mail Distribution

Richard H. Blount, II
Site Vice President
Surry Power Station
Virginia Electric & Power Company
Electronic Mail Distribution

Virginia State Corporation Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, VA 23209

Lillian M. Cuoco, Esq.
Senior Nuclear Counsel
Dominion Nuclear Connecticut, Inc.
Electronic Mail Distribution

Attorney General
Supreme Court Building
900 East Main Street
Richmond, VA 23219

Distribution w/encl:
G. Edison, NRR
RIDSNRRDIPMLIPB
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DATE	7/10/2002	7/11/2002	7/11/2002	7/12/2002	7/19/2002	7/ /2002	7/ /2002
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

U. S. NUCLEAR REGULATORY COMMISSION
REGION II

Docket Nos.:	50-280, 50-281
License Nos.:	DPR-32, DPR-37
Report Nos.:	50-280/02-07, 50-281/02-07
Licensee:	Virginia Electric and Power Company (VEPCO)
Facilities:	Surry Nuclear Power Station, Units 1 & 2
Location:	5850 Hog Island Road Surry, VA 23883
Dates:	June 10-14, 2002 and June 24-27, 2002
Inspectors	F. Jape, Senior Project Manager, Lead P. Fillion, Reactor Inspector R. Schin, Senior Reactor Inspector M. Thomas, Senior Reactor Inspector C. Fong, Intern (week of 6/10/02 only) R. Cortes, Intern
Approved by:	C. Payne, Acting Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000280-02-07, IR 05000281-02-07; Virginia Electric and Power Company; on 06/10-27/2002; Surry Nuclear Power Station, Units 1 & 2; Safety System Design and Performance Capability biennial baseline inspection, Steam Generator Tube Rupture.

This inspection was conducted by a team of regional engineering inspectors. No findings of significance were identified during this inspection. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversite Process, Revision 3, dated July 2000."

A. Inspection Identified Findings

None.

B. Licensee Identified Violations

None.

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Needs

a. Inspection Scope

A list of specific documents reviewed during this inspection can be found in the attachment to this report.

Instrumentation and Controls

The team reviewed surveillance procedures and completed tests for selected controls and indicators in the control room, that operators would use to change the state of pumps and valves needed to respond to a steam generator tube rupture (SGTR) event, to determine if they were consistent with Technical Specification (TS) and design requirements. The team also inspected the selected controls and indicators for appropriate human factors; such as labeling, arrangement, and visibility. The selected controls and indicators included switches for a volume control tank isolation valve, a charging pump suction valve from the refueling water storage tank, a pressurizer auxiliary spray valve, a main steam trip valve, a main steam non-return valve, a pressurizer power operated relief valve (PORV), a controller for a steam generator power operated relief valve, and an indicator for reactor coolant system subcooling.

Energy Source

The team reviewed the relevant design documents to determine if adequate voltage would be available to operate the steam generator PORVs, the steam generator blowdown trip valves, and the main feedwater isolation valve. The team also reviewed the application of overcurrent protective devices in these circuits from the viewpoint of spurious operation of the devices.

Controls

The team inspected the instruments listed below giving particular attention to the attributes of display and recording, labels, sensor location, range, TS requirements and power supply reliability.

- Condenser air ejector radiation monitor
- Steam generator blowdown radiation monitor
- Main steam line radiation monitor

- Turbine driven auxiliary feedwater pump exhaust radiation monitor
- Steam generator narrow range level
- Steam generator steam pressure
- Core exit thermocouples.

The team reviewed the basis for and control of the alert and alarm set points for the above listed radiation monitors. The team evaluated the loop uncertainty calculation for the steam generator narrow range level against standard methodology.

The method of inspection was a combination of review of relevant design documents, interviews with cognizant engineers and examination of installed instruments. The radiation monitors were examined during a walkdown inspection for location and material condition. In addition, radiation monitor computer stored data for selected dates or times were reviewed.

Operator Actions

The team reviewed selected portions of procedures that operators would use in identifying or responding to a steam generator tube leak or tube rupture including: emergency operating procedures (EOPs), abnormal procedures (APs), annunciator response procedures (ARPs), and operating procedures (OPs). The team also reviewed the SGTR mitigation strategy in the Updated Final Safety analysis Report, the Westinghouse Owners' Group Emergency Procedure Guidelines for SGTR, the licensee's EOP Step Deviation Document and the EOP Writers' Guide to determine if the procedures were consistent with these documents. The team also examined operator training lesson plans to assess if they were consistent with the procedures and the plant design. In addition, the team reviewed the list of operator workarounds to identify if any would affect operator actions during a SGTR event.

Heat Removal

The team reviewed the availability and reliability of the subsystems and equipment required to remove heat during a SGTR event as specified in licensing and design basis documents. The subsystems and equipment reviewed included the auxiliary feedwater (AFW) pumps, charging pumps, chemical and volume control system (CVCS), and safety injection system. This review included drawings, surveillance and operating procedures, test documentation, installed equipment and maintenance work orders.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

Installed Configuration

The team performed selective field inspections of the AFW pump room and the steam supply valves to the turbine driven AFW pump. Particular attention was placed on verifying that valves and components were in their required position and were consistent with design drawings. The purpose of these inspections was to assess the adequacy of the material condition and installation configurations.

Testing

The team reviewed periodic testing procedures and recent test results for selected equipment to determine if the equipment relied upon for mitigation of a SGTR event satisfied design analysis and TS requirements.

For at least one example of each of the below listed instruments, the team reviewed data sheets for the last two functional tests and calibrations. The criteria for this review was that the functional test and calibrations were performed at intervals specified by the TS and that any out-of-tolerance measurements or anomalies were addressed within the test procedure or following completion of the procedure. The instruments inspected were:

- Condenser air ejector radiation monitor
- Steam generator blowdown radiation monitor
- Main steam line radiation monitor
- Turbine driven auxiliary feedwater pump exhaust radiation monitor
- Steam generator narrow range level
- Steam generator steam pressure
- Pressurizer level
- Refueling water storage tank level

Automatic and operator control of the steam dump valves was reviewed to determine whether use of this system in normal plant shutdown demonstrates the modes of operation that would be used in response to a SGTR event.

The team reviewed surveillance procedures and test results for risk significant valves and pumps in the AFW, CVCS, and safety injection systems to verify compliance with TS, design basis requirements and inservice testing program requirements.

Operation

The team walked down selected portions of the EOPs, APs, ARPs, and Ops that operators would use in identifying or responding to a steam generator tube leak, or tube rupture. The team also observed the operators' use of the procedures during a simulator exercise of a SGTR event. During the walkdown and simulator exercise, the team assessed procedural correctness, operator knowledge, and human factors design of the procedures and related equipment against Nuclear Regulatory Commission (NRC) requirements for the quality of EOPs. The team also observed material conditions in the plant and simulator fidelity with the plant as needed to support effective operator training on EOPs.

RCS Leakage

The team reviewed recent reactor coolant system leakage monitoring results to determine if there were any early clues of a potential for an SGTR event.

b. Findings

No findings of significance were identified.

.3 Selected Components

a. Inspection Scope

Component Inspection

The team performed field inspections and reviewed maintenance, testing, and vendor documentation for selected components to assess the licensee's actions to verify and maintain the safety function, reliability, and availability of the components. The selected components included pumps, radiation monitors, main steam power operated relief valves, main steam safety valves, and check valves.

b. Findings

No findings of significance were identified.

.4 Identification And Resolution Of Problems

a. Inspection Scope

The team reviewed Plant Issues, industry operating experience reviews, and corrective actions related to SGTR events for the past five years to assess the adequacy of corrective actions.

The team reviewed a sample of problems identified by the licensee which were in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report. Inspection

Procedure 71152, “Identification and Resolution of Problems,” was used as guidance to perform this part of the inspection.

In addition, the team reviewed work orders on risk significant equipment to evaluate failure trends. The team also verified that the licensee was identifying procedural deficiencies at an appropriate threshold, was entering the deficiencies into the corrective action program, and that corrective actions were being taken for the identified deficiencies.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings

Exit Meeting Summary

On June 27, 2002, the team leader presented the inspection results to Mr. B. Foster and other members of licensee management and staff at the conclusion of the onsite inspection. The licensee's management acknowledged the findings presented.

The licensee’s representatives were aware that some proprietary information had been reviewed by the team, however, no proprietary information is contained in this report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee
M. Adams, Site Engineering Manager
J. Ashley, Licensing Engineer
A. Bagus, I&C Design
P. Blount, Radiation Protection
B. Foster, Director, Nuclear Safety and Licensing
K. Groves, Simulator Supervisor
E. Shore, Nuclear Engineering Supervisor
B. Sloan, Nuclear Engineering Supervisor
M. Thomas, System Engineer
W. Webster, System Engineer

NRC
G. McCoy, Resident Inspector
R. Musser, Senior Resident Inspector
C. Payne, Branch Chief, Acting, Region II

ITEMS OPENED, CLOSED, OR DISCUSSED

None

LIST OF DOCUMENTS REVIEWED

Procedures

0-HSP-LKRATE-001, Primary to Secondary Leak Rate Assessment Using Condenser Air Ejector Sample Data, Rev. 2
0-OSP-RC-002, Steam Generator Primary to Secondary Leakage Monitoring, Rev. 4
1-AP-10.05, Loss of Semi-Vital Bus, Attachment 3, Local Operation of SG PORVs, Rev. 13
1-AP-16.00, Excessive RCS Leakage, Rev. 9
1-AP-24.00, Minor SG Tube Leak, Rev. 6
1-AP-24.01, Large Steam Generator Tube Leak, Rev. 8
1-E-0, Reactor Trip or Safety Injection, Rev. 44
1-E-3, Steam Generator Tube Rupture, Rev. 23
1-ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery, Rev. 22
1-ES-0.1, Reactor Trip Response, Rev. 27
1-ES-3.1, Post-SGTR Cooldown Using Backfill, Rev. 15
1-OP-BD-001, Steam Generator Blowdown System Operation, Rev. 4
1-OP-32B, Steam Generator Blowdown System Alignment - Aux. & Turbine Bldgs., Rev. 2
1-RMA-A2, Unit 1 Mn Stm ABC Mon Alert/Hi, Rev. 4
1-RM-G8, Cndsr Air Ejctr Alert/Failure, Rev. 5
1-RM-H8, 1-SV-RI-111 High, Rev. 4
1-RM-N7, Stm Gen BD Alert/Failure, Rev. 2
1-RM-P7, 1-SS-RI-112 High, Rev. 1
CH-87.201, Primary to Secondary Leak Rate Monitoring, Rev. 4
CH-87.203, Primary to Secondary Leak Rate Calculation Using Tritium Analysis, Rev. 1
EPIP-1.01, Emergency Action Level Table, Reactor Coolant System Event, Rev. 43
OC-52, Identifying Increased RCS Leakage & Evaluation of Through-Wall Leakage on ASME Code Class Components, Rev. 05-29-02
VPAP-0505, Procedure Writers' Guide, Rev. 6
HP-3010.040, Radiation Monitoring System Setpoint Determination, Rev. 12
VPAP-3002, Operating Experience Program, Rev. 9
1-AP-21.01, Response to AFW Check Valve Back leakage, Rev. 2
1-GOP-2.5, Unit Cooldown, 351 ^{o}F to Less Than 205 ^{o}F, Rev. 15
1-OP-FW-001A, Auxiliary Feedwater System Alignment, Rev. 2
1-OPT-MS-005, Inservice Test of Main Steam Non-Return Valves, Rev. 1
0-MCM-0417-01, Velan Swing Check Valves Inspection and Overhaul, Rev. 7
SSES-8.09, Controlling Procedure for Component Engineering Check Valve Program, Rev. 3

Drawings

11448-FM-066A, Sheet 2, Flow/Valve Operating Numbers Diagram Aux Steam & Air Removal System, Rev. 42 (condenser air ejectors)
11448-FM-082D, Sheet 1, Flow/Valve Operating Numbers Diagram Sampling System, Rev. 16 (steam generator blowdown radiation monitor)
11448-FM-064A, Sheet 1, Flow/Valve Operating Numbers Diagram Main Steam System, Rev. 46 (main steam radiation monitor and steam generator pressure)

11448-FM-068A, Sheet 1, Flow/Valve Operating Numbers Diagram Feedwater System, Rev. 56 (steam generator narrow range level and feedwater isolation valve)
11448-FM-086B, Sheet 1, Flow/Valve Operating Numbers Diagram Reactor Coolant System, Rev. 28 (pressurizer level and PORV)
11448-FE-4BY, Wiring Diagram Effluent Radiation Monitoring Panel, Rev. 6
11448-FE-4BZ, Wiring Diagram Effluent Radiation Monitoring Panel, Rev. 5
11448-FE-6K, Wiring Diagram Radiation Monitoring Cabinet 1-1 Terminal Blocks, Rev. 18
11448-FE-6K4, Wiring Diagram Radiation Monitoring Cabinet 1-1 Terminal Blocks, Rev. 4
11448-FE-6K6, Wiring Diagram Radiation Monitoring Cabinet 1-1 Ratemeters, Controllers and Recorders, Rev. 3
11448-FE-6K8, Wiring Diagram Radiation Monitors 1-SS-RM-112/113, Rev. 3
5965D14, Interconnecting Wiring Diagram Steam Generator #1 Narrow Range Level LT-1-474 and LT-1-475, Rev. 12
5965D30, Interconnecting Wiring Diagram Loop 1 Steam Break Protection Steam Pressure PT-1-474, 475, 476, Rev. 12
5965D37, Interconnecting Wiring Diagram Pressurizer Level Protection Channel 1 and Level Control System, Rev. 15
5656D99, Steam Dump Control System Block Diagram, Rev. 3
5965D63, Interconnecting Wiring Diagram Steam Dump Control System, Rev. 9
11448-FE-18L, Wiring Details Miscellaneous Circuits, Rev. 36
11448-FE-18J, Wiring Details Miscellaneous Circuits, Rev. 23
11448-FE-6L, Wiring Diagram Radiation Monitoring Cabinet 1-2 Terminal Blocks, Rev. 12
11448-FE-6L6, Wiring Diagram Radiation Monitoring Cabinet 1-2 Ratemeters, Rev. 3
11448-FE-6L9, Wiring Diagram Radiation Monitors 1-SV-RM-111, Rev. 3
11448-FE-6N, Wiring Diagram High Range Effluent Radiation Monitors, Rev. 4
11448-FE-6S, Wiring Diagram High Range Effluent Radiation Monitors, Rev. 3
11448-FM-064A, Sheet 1, Main Steam System, Rev. 46
11448-FM-064A, Sheet 2, Main Steam System, Rev. 45
11448-FM-064A, Sheet 3, Main Steam System, Rev. 49
11448-FM-064A, Sheet 4, Main Steam System, Rev. 45
11448-FM-068A, Sheet 1, Feedwater System, Rev. 56
11448-FM-068A, Sheet 3, Feedwater System, Rev. 43
11448-FM-068A, Sheet 4, Feedwater Emergency Make-Up System, Rev. 26
11448-FM-088B, Sheet 1, Chemical and Volume Control System, Rev. 34
11448-FM-088B, Sheet 2, Chemical and Volume Control System, Rev. 39
11448-FM-088C, Sheet 1, Chemical and Volume Control System, Rev. 22
11448-FM-089A, Sheet 1, Safety Injection System, Rev. 55
11448-FM-089A, Sheet 2, Safety Injection System, Rev. 51
11448-FM-089A, Sheet 3, Safety Injection System, Rev. 47
11448-FM-089B, Sheet 4, Safety Injection System, Rev. 20
11448-FM-124A, Sheets 2&3, Steam Gen Blowdown Recirculation & Transfer System, Rev. 30

Completed Functional Tests and Calibrations

1-PT-26.2A, Radiation Monitoring Equipment Test (Victoreen Process Monitors) - functional test on 1-SV-R1-111 performed on 2/20/02 and 5/15/02
0-IPM-RM-G-001, Digital Ratemeter Model 942B Process Monitor Calibration, performed on 1-SV-R1-111 on 12/13/01
1-PT-26.2A, Radiation Monitoring Equipment Test (Victoreen Process Monitors) - functional test on 1-SS-R1-112 and -113 performed on 2/14/02 and 5/15/02
0-IPM-RM-G-001, Digital Ratemeter Model 942B Process Monitor Calibration, performed on 1-SS-R1-112 on 5/2/01
1-PT-26.2C, Radiation Monitoring Equipment Test (NRC Monitors) - functional test on 1-MS-RM-124, -125 and 126 on 2/12/02 and 5/9/02
CAL-260, NRC Radiation Monitor Calibration Models TA-600 & TA-900, performed on 1-MS-RM-124, -125 and -126 on 11/15/01
1-PT-26.2C, Radiation Monitoring Equipment Test (NRC Monitors) - functional test on 1-MS-RM-129 on 2/8/02
CAL-260, NRC Radiation Monitor Calibration Models TA-600 & TA-900, performed on 1-MS-RM-129 on 10/22/01
1-IPT-FT-FW-L-474, Steam Generator Level Protection Loop L-474 Functional Test, performed on 1/17/02 and 4/19/02
1-IPT-FT-FW-L-484, Steam Generator Level Protection Loop L-484 Functional Test, performed on 1/17/02 and 4/19/02
1-IPT-FT-FW-L-494, Steam Generator Level Protection Loop L-494 Functional Test, performed on 1/17/02 and 4/19/02
1-IPT-CC-FW-L-474, Steam Generator Level Protection Loop L-474 Channel Calibration, performed on 1/24/01 and 8/30/01
1-IPT-CC-FW-L-484, Steam Generator Level Protection Loop L-484 Channel Calibration, performed on 4/17/00 and 8/30/01
1-IPT-CC-FW-L-494, Steam Generator Level Protection Loop L-494 Channel Calibration, performed on 2/4/00 and 8/30/01
1-IPT-FT-MS-P-474, Steam Generator Pressure Loop P-474 Functional Test, performed 2/18/02 and 5/21/02
1-IPT-FT-MS-P-484, Steam Generator Pressure Loop P-484 Functional Test, performed 2/18/02 and 5/21/02
1-IPT-FT-MS-P-494, Steam Generator Pressure Loop P-494 Functional Test, performed 2/18/02 and 5/21/02
1-IPT-CC-MS-P-474, Steam Generator Pressure Loop P-474 Channel Calibration, performed on 2/13/00 and 9/12/01
1-IPT-CC-MS-P-484, Steam Generator Pressure Loop P-484 Channel Calibration, performed on 2/13/00 and 9/12/01
1-IPT-CC-MS-P-494, Steam Generator Pressure Loop P-494 Channel Calibration, performed on 2/13/00 and 9/12/01
1-IPT-FT-RC-L-459, Pressurizer Level Protection Loop L-459 Functional Test, performed on 1/17/02 and 4/19/02
1-IPT-CC-RC-L-459, Pressurizer Level Protection Loop L-459 Channel Calibration, performed on 4/29/00 and 10/22/01

1-IPT-FT-CS-L-100A, Refueling Water Storage Tank Level Loop L-100A Functional Test, performed 1/17/02 and 4/19/02
1-IPT-CC-CS-L-100A, Refueling Water Storage Tank Level Loop L-100A Channel Calibration, performed on 3/25/00 and 9/02/01
1-IPT-CC-RC-ICCM-001, Inadequate Core Cooling Monitor Train A Calibration, Rev. 8, completed 10/25/01
1-OPT-RC-001, PRZR PORV Refueling Test, Rev. 9, completed 11/6/01
1-OPT-SI-003, Quarterly Test of SI MOVs and RWST Crosstie TVs, Rev. 12, completed 4/8/02
1-OPT-SI-020, CSD Test of Charging and Safety Injection MOVs and Check Valves, Rev. 3, completed 11/15/02
1-OSP-FP-008, Appendix R Fail-Safe Valve Actuation Test, Rev. 0, completed 3/23/97
1-OSP-FP-008, Appendix R Fail-Safe Valve Actuation Test, Rev. 2, completed 5/5/00
1-PT-14.2, Main Steam Trip and Non-Return Valve Operability Verification, Rev. 6, completed 12/5/01
1-PT-14.5, Test of Main Steam Power Operated Relief Valves, Rev. 8, completed 2/24/02 and 6/1/02

Completed Work Orders

WO 0042290301, Slight Packing Leak on Westside of Trip Valve, dated 05/06/00
WO 0042560401, SG-1C AFW Pump 2 Steam Supply Isolation Valve 01-MS-158, dated 04/18/00
WO 0043889701, Check Source is Sticking on the Cover, dated 02/08/01
WO 0044205101, SG-1A Header Safety Valve 01-MS-SV-101A, dated 11/08/01
WO 0044205201, SG-1B Header Safety Valve 01-MS-SV-101B, dated 11/08/01
WO 0044205401, SG-1A Header Safety Valve 01-MS-SV-102A, dated 11/08/01
WO 0044205801, SG-1B Header Safety Valve 01-MS-SV-103B, dated 11/08/01
WO 0044446101, VOTES Testing for SG-1C Main Steam Non-Return Valve 01-MS-NRV-101C, dated 10/14/01
WO 0044782701, AFW Pump 2 Turbine Trip Valve Preventive Maintenance, dated 05/25/01
WO 0045440201, 3/8" Air Supply Tubing to Trip Valve Bent Multiple Times, dated 10/31/01
WO 0045695501, AFW Pump 2 Turbine Trip Valve Preventive Maintenance, dated 10/20/01
WO 0045981101, Insulation Brittle and Cracked on Detector Cable, dated 11/14/01
WO 0046080401, Alarm Coming in Occasionally with no Problem Evident and no Monthly Check Source in Progress, dated 12/05/01
WO 0046121601, Radiation Monitor Failure Light & Alarm Coming in and Clearing, RM is inoperable, dated 12/10/01

Calculations

EE-0432, Channel Statistical Allowance Calculations for Surry Power Station, Units 1&2, Loops 1474, 1475, 1476, etc., Rev. 1, dated 10/26/93
07797.06-E-001, 125 VDC Voltage Drop Calculation for Selected Safety Related and Non-safety Related Components, Rev. 0, dated 6/28/00
CAL-260, NRC Radiation Monitor Calibration Models TA-600 and TA-900, Rev. 11
CAL-817, Model 942 Log Ratemeter Scintillation Detector Source Calibration, Rev. 22

Vendor/Technical Manuals

V659-00012, Victoreen Instruction Manual Gamma Detectors Models, Rev. 2
N001-00004, Operation & Maintenance Manual TA-900 Area Radiation Monitoring System, Rev. 1

Westinghouse Owners' Group (WOG) Emergency Guidelines and Surry Step Deviation Documents (SDD)

SDD E-0, Reactor Trip or Safety Injection, Rev. 2-20-2002
SDD E-3, Steam Generator Tube Rupture, Rev. 1-7-2002
SDD ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired
WOG E-0, Reactor Trip or Safety Injection, Rev. 1C
WOG E-3, Steam Generator Tube Rupture, Rev. 1C
WOG ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired, Rev. 1C

Operator Training Lesson Plans

ND-88.1-LP-6, Abnormal Procedure AP-16, Excessive RCS Leakage, Rev. 16
ND-95.3-LP-13-DRR, E-3, Steam Generator Rube Rupture, Rev. 11
ND-95.3-LP-23, Emergency Response Guidelines - ECA-3.1, SGTR With LOCA - Subcooled Recovery, Rev. 10
RQ-00.5-TS-5, AP-16.00 Modifications, Rev. 0
RQ-00.6-ST-1-DRR, Small and Large SG Tube Leaks, Rev. 6
RQ-01.2-LP-6-DRR, EPRI Guidelines for Small Generator Tube Leaks, Rev. 0

UFSAR

UFSAR Section 10.3.1, Main Steam System, Rev. 33
UFSAR Section 11.3.3, Process Radiation Monitoring System, Rev. 33
UFSAR Section 14.3.1.1, Steam Generator Tube Rupture, Rev. 33

Completed Performance/Surveillance Test Procedures

0-MCM-0427-01, Main Steam Safety Valve Removal and Installation, Rev. 6, dated 11/08/01
1-MPT-0427-02, Main Steam Safety Setpoint Verification, Rev. 8, dated 11/09/01
1-OPT-CH-002, Charging Pump Operability and Performance Test for 1-CH-P-1B, Rev. 27, dated 02/11/02, and 03/28/02
1-OPT-FW-001, Motor Driven Auxiliary Feedwater Pump 1-FW-P-3A, Rev. 8, dated 05/05/01, 06/04/01, 08/27/01, 10/31/01, 11/19/01, and 12/03/01
1-OPT-FW-003, Turbine Driven Auxiliary Feedwater Pump 1-FW-P-2, Rev.15, dated 05/07/01, 05/25/01, 08/13/01, 11/08/01, 12/06/01, 12/07/01, 12/08/01
1-OPT-SI-002, Refueling Test of the Low Head Safety Injection Check Valves to the Cold Legs, Rev. 9, dated 10/23/01
1-OPT-SI-014, Cold Shutdown Test of the Safety Injection Check Valves to RCS Hot and Cold Legs, Rev. 6, dated 12/01/01

1-OPT-SI-007, Refueling Test of the High Head Safety Injection Check Valves to the Cold Legs, Rev. 11, dated 10/29/01
1-OPT-SI-012, Refueling Test of Low Head Safety Injection Lines to Charging Pumps, Rev.12, dated 10/31/01
1-OSP-SI-002, Charging Pump Head Curve Verification, Rev. 2, dated 10/20/01
2-OPT-SI-002, Refueling Test of the Low Head Safety Injection Check Valves to the Cold Legs, Rev. 8, dated 04/10/02
2-OPT-SI-014, Cold Shutdown Test of the Safety Injection Check Valves to RCS Hot and Cold Legs, Rev. 11, dated 04/15/02

PIs Issued as a Result of this Inspection

PI-S-2002-2244	3^{rd} continuous action in E-0 for SI initiation includes 30 deg F subcooling, but step 4 of E-0 does not have the same criteria.
PI-S-2002-2244	AP 24.01 appears to take precedence over ES 0.1. The transition from EOPs to AP 24.01 is a NOTE (not in an action statement) and does not use the words..."go to". This is not in agreement with the Writers Guide.
PI-S-2002-2173	In a simulator exercise, operators relied on HP for taking local surveys on main steam lines to identify which SG had the rupture. However, HP personnel on shift said they would not take such local surveys.
PI-S-2002-2129	During the simulator demonstration, the steam generator blowdown radiation monitor did not indicate a leak. Is this a simulator problem?
PI-S-2002-2130	During the simulator demonstration, the main steam radiation monitors did not reflect actual plant condition. They indicated high radiation from one SG early in the event, however the actual in-plant main steam rad monitors would not give this early indication of a SG leak. Is this a simulator problem?
PAR issued	Procedure 1-OP-RM-001, for realigning the two blowdown rad monitors to the three SGs, requires local operator action that is difficult to accomplish due to location and should be completed quickly. The stated action may not be needed since there are alternate means of accomplishing the action.
PI-S-2002-2122	The auxiliary feedwater steam turbine has two steam exhaust lines, but the radiation monitor (½ MS-RM-129 (229), is physically connected to one of the lines. With this arrangement is the calculation of activity released affected.
PI-S-2002-2144	The main steam safety valves 101 A, B and C are classified as Maintenance Rule a(4) components, but valves 102-105 A, B and C are not.
PI-S-2002-2182	What is the basis for the main steam radiation monitor setpoint and the steam driven auxiliary feedwater pump radiation monitor setpoint?
PI-S-2002-2123	The outer jacket of the electrical cable for 01-MS-RM-129 at the detector location is cracked. What effect does this have on electrical integrity?
PI-S-2002-2215	Completed procedure 1-IPT-FT-FW-L-484, SG level protection loop, identified the as found comparator LC-1-148 output #2 was outside the allowed tolerance, but no as left value was recorded in the procedure.

PI-2002-2249 Shift Technical Advisors could not call up the recorded information (two-hour storage time) for main steam and steam driven auxiliary feedwater turbine exhaust radiation monitors as required by RG 1.97.

Operational Events Reviewed

S-1995-0205-E1, Diagnosis and Mitigation of RCS Leakage Including SGTRs
S-1993-0124-E1, Weakness in EOPs Found as a result of SGTR
S-1994-0134-E1, Operational Experience On SGTRs and Leaks
S-1997-3653-E1, Main Steam Range Radiation Monitor Inoperable Due to Equipment Failure
S-1997-3857-E1, UFSAR Time Requirement For Terminating S/G Tube Rupture Flow Not Met During Simulator Training
S-1996-3191-E1, Plant Event-SGTR
S-1998-4405-E1, Radiation Monitor Baseline Data Trending
S-1996-3237-E1, Long Term Inoperability of Both Pressurizer PORVs
S-1998-0230-E1, Problems Experienced During a SGTR
S-1997-3896-E1, Single Failure of a Power Supply Could Limit Ability to Cool Down and Depressurize Within UFSAR Time Limit
S-2000-1727-E1, Steam Generator Tube Failure
S-1998-0036-E1, North Anna S/G Tube Rupture Event
S-1998-4142-E1, Excessive Operator Response Times Due to Inadequate Analysis Implementation and 3-Legged Communication

Design Basis Documents

System Design Basis Document SDBD-SPS-AFW, Auxiliary Feedwater System, Rev. 3
System Design Basis Document SDBD-SPS-CH, Chemical and Volume Control System, Rev. 0
System Design Basis Document SDBD-SPS-MS, Main Steam and Ancillary Systems, Rev. 1
System Design Basis Document SDBD-SPS-SI, Safety Injection System, Rev. 3